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NUREG-2246

FUEL QUALIFICATION FOR ADVANCED REACTORS

~~Draft Report for Comment~~

Office of Nuclear Reactor Regulation

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~~Draft Report for Comment~~

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ABSTRACT

Proposed advanced reactor technologies~~designs~~ use fuel designs and operating environments (e.g., neutron energy spectra, fuel temperatures, neighboring materials) that differ from traditional light-water reactor fuel for which there is a considerable~~the large~~ experience base ~~available for traditional light-water reactor fuel~~. The purpose of this report is to identify criteria that will be useful for advanced reactor designers through an assessment framework that would support regulatory findings associated with nuclear fuel qualification. The report examines~~begins by examining~~ the regulatory basis and related guidance applicable to fuel qualification, noting that the role of nuclear fuel in the protection against the release of radioactivity for a nuclear ~~facility~~reactor depends heavily on the reactor design. The report considers the use of accelerated fuel qualification techniques and lead test specimen programs that may shorten the timeline for qualifying fuel for use in a nuclear reactor at the desired parameters (e.g., burnup). The assessment framework particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate evaluation models and empirical safety criteria.

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ABBREVIATIONS AND ACRONYMS

AFQ	accelerated fuel qualification
AOO	anticipated operational occurrence
ARDC	advanced reactor design criterion/criteria
ATR	advanced test reactor
ED	experimental data
EM	evaluation model
FAST	fission accelerated steady-state test
FQAF	fuel qualification assessment framework
G	goal
GDC	general design criterion/criteria
GESTAR	General Electric standard application for reactor fuel
LWR	light-water reactor
OBE	operating basis earthquake
PCMI	pellet-clad mechanical interaction
PCMM	predictive capability maturity model
PDC	principal design criterion/criteria
PIRT	phenomena identification and ranking table
SAFDL	specified acceptable fuel design limit
SARRDL	specified acceptable radionuclide release design limit
SSC	structure, system, and component
SSE	safe shutdown earthquake
TREAT	transient reactor test facility
TRISO	tristructural-isotropic
U-10Zr	uranium alloy with 10 weight percent zirconium
U-Pu-10Zr	uranium-plutonium alloy with 10 weight percent zirconium
UO_2	uranium dioxide

1 INTRODUCTION

1.1 Purpose

The purpose of this report is to provide a fuel qualification assessment framework for use with advanced reactor designs, including power and non-power reactors, that addresses regulatory requirements. Specifically, the framework provides criteria (referred to as base goals[1]) derived from regulatory requirements that, when satisfied, would support regulatory findings that a nuclear fuel is qualified. This report provides the bases for the identified base goals and clarifying examples for the types of information that an applicant should provide for the NRC to determine that these goals are satisfied, and regulatory requirements are met. Appendix A lists all goals within the framework.

This framework relies on regulatory requirements that are applicable to applications for design certifications, combined licenses, manufacturing licenses, or standard design approvals. While the requirements of Title 10 of the Code of Federal Regulations (10 CFR) 50.43(e) are not applicable to applications for a construction permit, the remaining requirements, identified in Section 2.1 of this report, are generically applicable to power reactor applications. Accordingly, the framework provides applicants with criteria for satisfying regulatory requirements for applications for a design certification, combined license, manufacturing license, standard approval, and for the development of a fuel qualification plan to support a construction permit application.

1.2 Definitions

The term “fuel qualification” is not explicitly defined or used in NRC regulations. However, there are regulatory requirements applicable to power reactor applications that are generally associated with nuclear fuel behavior under conditions of normal operation, anticipated operational occurrences (AOOs), and accident conditions (see Section 2.1 of this report). The information needed to understand fuel behavior under these conditions is generally obtained as part of a fuel qualification process. Research literature provides additional insight into the objectives of fuel qualification from the developers’ points of view (Crawford, et. al., 2007), (Terrani, et. al., 2020) which highlight the needs to (1) fabricate~~ion~~ a fuel product in accordance with a specification, (2) meet licensing safety-requirements, and (3) meet reliability needs. This report uses the following definitions of “qualified fuel” and “fuel qualification” as an aid to NRC staff in the development of the fuel qualification framework.~~because they characterize regulatory needs based on NRC regulations, staff’s experience form licensing solid fuel designs (particularly light water reactor (LWR) fuel), advanced reactor fuel testing performed to-date, and AFQ considerations.~~

> “Qualified fuel” means fuel for which reasonable assurance exists that the fuel, fabricated~~ion~~ in accordance with its specification, will perform as described in the safety analysis.

> “Fuel qualification” means the overall process (planning, testing, analysis, etc.) used to obtain qualified fuel.

[1] A base goal is a goal that is not decomposed any further but is supported by evidence. The top-down approach and decomposition of regulatory requirements into base goals is described in Section 3 of this report.

These definitions were used because they characterize information required by NRC regulations as described in Section 2.1, “Regulatory Basis” and are consistent with NRC staff’s experience from licensing solid fuel designs (particularly light water reactor (LWR) fuel), advanced reactor fuel testing performed to-date, and AFQ considerations.

1.3 Safety Case

The role of nuclear fuel in the protection against the release of radioactivity can vary depending on the reactor design[2]. For example, ~~facilities~~nuclear power plants that use traditional oxide fuels with metal cladding are designed with robust barriers (e.g., containment buildings) to prevent the release of radioactive material under postulated accident conditions, whereas a ~~facility~~ plant that uses tristructural-isotropic (TRISO) fuel may credit a series of barriers (including barriers within the fuel itself) to prevent the release of radioactive material (i.e., a functional containment (NRC, 2018a)). Thus, in the nuclear fuel qualification process, it is essential to specify the fission product retention functions of the nuclear fuel (this is addressed under Goal (G) 2, “Safety Criteria,” in Section 3.2 of this report).

1.4 Scope

Nuclear fuel affects many aspects of nuclear safety, including neutronic performance (e.g., reactivity feedback), thermal-fluid performance (e.g., margin to critical heat flux limits), fuel mechanical performance, reactor core seismic behavior, fuel transportation, and storage. The scope of this report focuses on the identification and understanding of fuel life-limiting failure and degradation mechanisms due to irradiation and irradiation assisted phenomena during reactor operation. The assessment criteria in Section 3 (referred to as goals) of this report draw on regulatory experience gained from licensing solid fuel reactor designs (particularly LWR designs), results from advanced reactor fuel testing performed to-date, and accelerated fuel qualification (AFQ) considerations. An attempt has been made to develop a generically applicable set of base goals~~criteria~~. However, some ~~criteria~~goals may not apply to all fuel types or reactor designs ~~liquid fuel forms~~ (e.g., molten salt reactor fuel), and ~~these fuel forms may require~~ additional or alternate critera~~riteria~~ should be applied (see Section 2.2.4 of this report for guidance on molten salt reactor fuel).

[2] Fuel qualification literature often use the term “safety case”. This term is undefined but generally refers to the safety functions that the fuel is relied upon to perform. Principally among these safety-functions is the protection against the release of radionuclides.

2 BACKGROUND

2.1 Regulatory Basis

Nuclear fuel qualification to support reactor licensing involves the development of a basis to support findings associated with regulatory requirements that apply to ~~the nuclear facility.~~nuclear reactors. This section discusses these requirements and their relationship to this report.~~ Note that satisfying the fuel qualification framework criteria only "partially addresses" the requirements associated with the nuclear facility~~reactor~~. This is because t~~ The fuel qualification framework provides a means to identify the safety criteria only for the fuel and it is the safety criteria for the fuel that establish the performance criteria for some structures, systems, and components (SSCs) of the facility. Therefore, addressing the criteria in the fuel qualification framework provides the information necessary to meet the regulations associated with fuel qualification, but does not in and of itself satisfy regulatory requirements. The requirements are fully addressed through the description and analysis of these SSCs in an application.

The relevant regulatory requirements associated with fuel qualification are as follows:

- The regulation in 10 CFR 50.43(e)(1)(i) requires demonstration of the performance of each safety feature[3] of the design through either analysis, appropriate test programs, experience, or a combination thereof. The assessment framework developed in Section 3 of this report (1) provides a means to identify the safety features of the fuel necessary to comply with regulatory requirements (see Goal (G) 2, "Safety Criteria," in Section 3.2 of this report), and (2) clarifies the types of evidence (e.g., analysis, testing, experience) typically expected to demonstrate these safety features. In accordance with the scope of this report, the safety features assessed in Section 3 of this report are associated with the identification and understanding of fuel life-limiting failure and degradation mechanisms that are due to irradiation and irradiation assisted phenomena during reactor operation.
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- The regulation in 10 CFR 50.43(e)(1)(iii) requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. This range appears in G2.1.1, "Definition of Fuel Performance Envelope," which is discussed in Section 3.2.1.1 of this report. Additionally, the evaluation model assessment framework in Section 3.3 provides criteria for assessing analytical tools, and the experimental data assessment framework in Section 3.4 provides criteria for data adequacy.
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- The regulations in 10 CFR 50.34(a)(1)(ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi) require an evaluation of a postulated fission product release. This requirement can be partially addressed by satisfying G2.2, "Radionuclide Release Limits," which is discussed in Section 3.2.2 of this report.
-

[3] Nuclear fuel contributes to the reactivity balane and is a source of heat generation and fission products. Therefore, nuclear fuel is generally recognized as impacting the safety functions of reactivity control, heat removal, and confinement of radioactive material.

The regulations in 10 CFR 50.34(a)(3)(i), 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a) require that the principal design criteria (PDC) be provided for a construction permit, design certification, combined license, standard design approval, or manufacturing license. Appendix A to 10 CFR Part 50, "General Design Criteria [GDC] for Nuclear Power Plants," establish the minimum requirements for PDC for water-cooled nuclear power plants. Appendix A to 10 CFR Part 50 also established that the GDC are generally applicable to other types of nuclear power units and are intended to provide guidance in determining the PDC for such other units. Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," (NRC, 2018b) provides guidance on how the GDC in Appendix A to 10 CFR Part 50 may be adapted for non-LWR designs and contains advanced reactor design criteria (ARDC). While the GDC and ARDC are not requirements for non-LWR designs, the GDC and ARDC identified below address safety functions generally associated with nuclear fuel that are not otherwise captured by NRC regulations (e.g., reactivity control, heat removal, confinement of radionuclides). Accordingly, NRC staff expects the following GDC and ARDC to be addressed as part of fuel qualification:

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    - GDC~~)~~ 2 and ~~Advanced Reactor Design Criterion~~[4] ~~(~~ARDC~~)~~ 2, "Design bases for protection against natural phenomena," ~~of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities,"~~ requires that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. Appendix S to 10 CFR 50, "Earthquake engineering criteria for nuclear power plants," implements GDC 2 as it pertains to seismic events and defines specific earthquake criteria for nuclear power plants. This appendix established definitions for safe shutdown earthquake (SSE), operating basis earthquake (OBE), and safety requirements for relevant SSCs. These SSCs are necessary to assure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe-shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures. The safety functions generally associated with nuclear fuel include control of reactivity, cooling of radioactive material, and confinement of radioactive material[5]. The requirements related to natural phenomena can be partially addressed by satisfying G2.3, "Safe Shutdown," which is discussed in Section 3.2.3 of this report.

~~[4] While the GDC and ARDC are not requirements for non-LWR designs, they are considered guidance for non-LWR advanced reactor applicants in developing proposed principal design criteria (PDCs). PDCs are required to be proposed by applicants. Regulatory Guide 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," (NRC, 2018b) provides guidance on how the GDC in Appendix A to 10 CFR Part 50 may be adapted for non-LWR designs.~~

[5] These "fundamental safety functions" are identified in the IAEA safety glossary (IAEA, 2018) and are also incorporated into NRC regulations. Reactivity control is specified by GDC 27 and ARDC 26; heat removal is specified by GDC/ARDC 10, GDC 27, ARDC 26, and GDC/ARDC 35; radionuclide retention is specified by GDC/ARDC 10 and is associated with the requirements under 10 CFR 50.34(a)(1)((ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi).

- GDC 10 and ARDC 10, “Reactor Design,” require that specified acceptable fuel design limits (SAFDLs) or specified acceptable radionuclide release design limits (SARRDLs) not be exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). This requirement can be partially addressed by satisfying G2.1, “Design Limits during Normal Operation and AOOs,” which is discussed in Section 3.2.1 of this report.

- GDC 27 and ARDC 26, “Combined Reactivity Control Systems Capability,” require, in part, the ability to achieve and maintain safe shutdown under postulated accident conditions and provide assurance that the capability to cool the core is maintained. This requirement can be partially addressed by satisfying G2.3, “Safe Shutdown,” which is discussed in Section 3.2.3 of this report.

- GDC 35 and ARDC 35, “Emergency Core Cooling,” require an emergency core cooling system that provides sufficient cooling under postulated accident conditions; they also require that fuel and clad damage that could interfere with continued effective core cooling is prevented. This requirement can be partially addressed by satisfying G2.3, “Safe Shutdown,” which is discussed in Section 3.2.3 of this report.

- ~~The regulations in 10 CFR 50.34(a)(1)(ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi) require an evaluation of a postulated fission product release. This requirement can be partially addressed by satisfying G2.2, “Radionuclide Release Limits,” which is discussed in Section 3.2.2.~~

~~GDCs and ARDCs are not requirements for non-LWR designs and are instead considered guidance for non-LWR advanced reactor applicants in developing proposed principal design criteria (PDCs). PDCs are required to be proposed by applicants in accordance with the following regulations: 10 CFR 50.34, “Contents of applications; technical information;” 10 CFR 52.47, “Contents of applications; technical information;” 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report;” 10 CFR 52.137, “Contents of applications; technical information;” and 10 CFR 52.157, “Contents of applications; technical information in final safety analysis report.” To the extent that an applicant determines that a proposed PDC associated with fuel design for their specific design is necessary to establish necessary design and performance requirements that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public, such a PDC may be deemed necessary by the NRC staff. Applicants for LWRs, including small modular LWRs, may be required to meet the GDC in Appendix A to 10 CFR Part 50 for fuel design and qualification. Certain GDC requirements associated with fuel performance (e.g. reactivity) are not addressed elsewhere in NRC regulations. The staff’s expectation is that these aspects will be addressed in a licensing application.~~ The fuel qualification assessment framework in Section 3 of this report provides guidance to facilitate an efficient and transparent licensing review in the area of fuel qualification. The guidance provided in this report is not a substitute for the Commission’s regulations, and compliance with the guidance is not required.

2.2 Related Guidance

Several guidance documents are available or are in development that ~~address~~ are related to nuclear fuel qualification. This section discusses these guidance documents and their relationship to this report.

2.2.1 NUREG-0800, Section 4.2

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 4.2, Revision 3, “Fuel System Design,” issued March 2007 (NRC, 2007), lists acceptance criteria that NRC staff considers in a licensing review for a LWR fuel system. Section 3.2 of this report captures the objectives of the fuel system safety review as follows[6]:

- Assurance that the fuel system is not damaged as a result of normal operation and AOOs can be demonstrated, in part, by meeting G2.1, “Design Limits during Normal Operation and AOOs,” which is discussed in Section 3.2.1 of this report.

- Assurance that fuel system damage is never so severe as to prevent negative reactivity insertion (e.g. control element insertion) when required can be demonstrated, in part, by meeting G2.3, “Safe Shutdown,” which is discussed in Section 3.2.3 of this report. Section 3.2.3.2 of this report discusses the specific item of negative reactivity~~control element~~ insertion.

- Assurance that the number of fuel ~~rod~~ failures is not underestimated for postulated accidents can be demonstrated, in part, by meeting G2.2, “Radionuclide Release Limits,” which is discussed in Section 3.2.2 of this report.

- Assurance that coolability is always maintained can be demonstrated, in part, by meeting G2.3, “Safe Shutdown,” which is discussed in Section 3.2.3 of this report. Section 3.2.3.1 of this report discusses the specific item of maintaining a coolable geometry.

NUREG-0800, Section 4.2, provides guidance regarding traditional LWR fuel and the licensing bases for traditional LWR power plants. Specifically, NUREG-0800, Section 4.2, evaluates fuel system designs for known fuel failure mechanisms from traditional LWR fuel (i.e., uranium dioxide (UO_2) fuel with zirconium-alloy cladding), identifies specific testing for addressing key LWR fuel phenomena, and includes empirical acceptance criteria based on testing of LWR fuel samples. As such, the specific acceptance criteria provided in NUREG-0800, Section 4.2, may not apply or may not suffice to address advanced reactor technologies that use different fuel forms, or address situations in which the fuel plays different roles in the protection against the release of radionuclides. However, this report incorporates lessons learned from the development of the acceptance criteria in NUREG-0800, Section 4.2, as follows:

- The significant effect of fuel manufacturing parameters on fuel performance is addressed through G1, “Fuel Manufacturing Specification,” which is discussed in Section 3.1 of this report.

- Limitations on test facilities (e.g. Advanced Test Reactor (ATR) and the Transient Reactor Test Facility (TREAT)) and the risks of obtaining irradiated fuel data are

[6] Neither NRC regulations nor NUREG-0800 uses the term “fuel qualification.” NUREG-0800, Section 4.2 provides acceptance criteria to support regulatory findings associated with fuel performance.

discussed in the experimental data assessment framework in Section 3.4 of this report and are also mentioned in Section 3.2.2.3.1 of this report.

2.2.2 ATF-ISG-2020-01

ATF-ISG-2020-01, “Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept,” issued January 2020 (NRC, 2020a), provides supplementary guidance to NUREG-0800, Section 4.2. The guidance was developed using a phenomena identification and ranking table (PIRT) process[7] and is specific to applications involving fuel products with chromium-coated zirconium alloy cladding. Like the guidance in NUREG-0800, Section 4.2, the specific phenomena identified in ATF-ISG-2020-01 may not apply to advanced reactor technologies. However, as discussed in the evaluation model assessment framework in Section 3.3 of this report, the PIRT process may be used as one acceptable method to identify failure mechanisms and necessary features of an evaluation model~~, as discussed in the evaluation model assessment framework in Section 3.3 of this report~~.

2.2.3 Regulatory Guide 1.233

Regulatory Guide 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors,” issued June 2020 (NRC, 2020b), provides guidance for a modern, risk-informed approach to licensing reviews by informing the licensing basis and determining an appropriate level of information for parts of preliminary or final safety analysis reports for advanced non-LWRs. This approach emphasizes assessing facility risk by quantifying event frequencies and the associated radiological consequences. The consequence evaluation aspect of the risk assessment is addressed, in part, by G2.2, “Radionuclide Release Limits,” which is discussed in Section 3.2.2 of this report.

Additionally, Regulatory Guide 1.233 discusses fundamental safety functions. Nuclear fuel contributes to the reactivity balance and is a source of heat generation and fission products. Therefore, nuclear fuel is generally recognized as impacting the fundamental safety functions of reactivity control, heat removal, and confinement of radioactive material. Fuel qualification may partially address~~es~~ the~~se~~ fundamental safety functions ~~of control of reactivity, cooling of radioactive material, and confinement of radioactive material~~ by incorporating the role of the fuel in these safety functions in G2, “Safety Criteria,” which is discussed in Section 3.2 of this report, as follows:

- Confinement of radioactive material is partially addressed by G2.1, “Design Limits during Normal Operation and AOOs,” and G2.2, “Radionuclide Release Limits.” These goals allow for a graded approach such that information should be provided in accordance with the degree to which the fuel is credited in the protection against the release of radionuclides.

- Control of reactivity and cooling or radioactive material are partially addressed by G2.3, “Safe Shutdown.” This goal allows for a graded approach such that information should

[7] See Regulatory Guide 1.203, “Transient and Accident Analysis Methodologies,” for more information on the PIRT process (NRC, 2005).

be provided in accordance with the degree to which the fuel can obstruct the insertion of negative reactivity.

2.2.4 Guidance in Development

The U.S. Nuclear Regulatory Commission (NRC) staff is currently developing guidance in additional areas related to fuel qualification. As discussed in Section 1.4 of this report~~3~~, the role of fuel in the protection against the release of radioactivity~~safety case~~ for reactors that use nonsolid fuel forms may require additional or alternative criteria to those in this report. To that end, the NRC is supporting the development of a proposed methodology for molten salt reactor fuel salt qualification (ORNL, 2018) (ORNL, 2020).

Additionally, G2, “Safety Criteria,” addresses the role of the fuel in the protection against the release of radioactivity, as discussed in Section 1.3 of this report~~2~~. G2 is supported by source term considerations, as detailed in G2.2.1, “Radionuclide Retention Requirements,” and G2.2.3, “Conservative Modeling of Radionuclide Retention and Release.” Furthermore, G2.1, “Design Limits during Normal Operation and AOOs,” discusses SARRDLs, which involve the use of a source term. The NRC is supporting the development of source term guidance for non-LWRs which may affect this aspect of fuel qualification (SAND, 2020) (INL, 2020).[8]

2.3 Accelerated Fuel Qualification

AFQ involves, in part, the use of advanced modeling and simulation to inform constituent and system selection and to enable integral fuel performance analyses (Terrani, et al., 2020). The AFQ process, shown in Figure 2-1, supports the identification of important parameters and phenomena for targeted characterization through separate-effects tests.

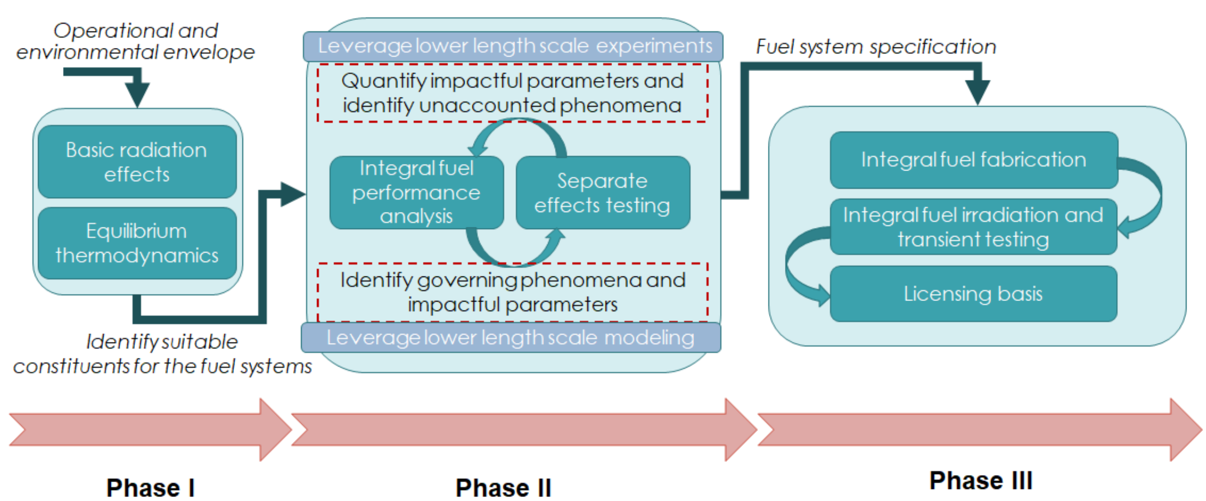

Figure 2-1 AFQ Process Workflow (Terrani, et al., 2020)

Advanced separate-effects testing techniques, such as fission accelerated steady-state testing (FAST) (Beausoleil II, Povirk, & Curnutt, 2020) and MiniFuel (Petrie, Burns, Raftery, Nelson, &

[8] The guidance developed on source term does not alter the fuel qualification framework. Both the guidance on source term and the fuel qualification framework accommodate a graded approach to source term where simplified, conservative models can be used to reduce the data requirements.

Terrani, 2019), can reduce the time needed to achieve a given burnup and provide basic data on material behavior and property evolution under irradiation conditions. The information obtained through these analyses and separate-effects tests could help justify the adequacy of the evaluation model as part of Evaluation Model (EM) G1, "Evaluation Model Capabilities," which is discussed in Section 3.3.1 of this report. Additionally, validated physics-based models may support some extrapolation of evaluation models beyond the limits of available integral test data, as noted under EM G.2.2.4, "Restricted Domain," in Section 3.3.2.2.4 of this report. Ultimately, the AFQ process relies on integral irradiation test data to validate engineering scale fuel performance codes and to confirm the performance and safety of the fuel system under prototypic conditions. Accordingly, the integral test data produced as part of the AFQ process appear to be consistent with the considerations in the experimental data assessment framework discussed in Section 3.4 of this report.

2.4 Lead Test Specimens

Much of the data necessary to qualify fuel for use come from irradiated test specimens. Lead test specimens have been successfully used in operating reactors to obtain data at the needed exposures and are discussed in NUREG-0800, Section 4.2, as well as in Section 3.4.2 of this report. Section 3.4.2 of this report further discusses the potential for use of lead test specimens beyond what has been traditionally used for LWRs that can be useful for advanced reactor technologies.

2.5 First Core Applications

Nuclear fuel contributes to the reactivity balance and is a source of heat generation and fission products. Therefore, it is generally recognized as impacting the safety functions of reactivity control, heat removal, and confinement of radioactive material. Accordingly, nuclear power reactor fuel is generally considered a safety feature subject to the requirements of 10 CFR 50.43(e) and would require demonstration prior to licensing.

Section 2.1 of this report identifies the requirements under 10 CFR 50.43(e)(1) to demonstrate the performance of each safety feature of a design and to have sufficient data on the safety feature to assess analytical tools. The assessment framework in Section 3 of this report is developed to address the requirements identified in Section 2.1 of this report, including 10 CFR 50.43(e)(1)(i) and 10 CFR 50.43(e)(1)(iii). The framework provided in Section 3 of this report accommodates the use of a lead test specimen program beyond what has been traditionally used for LWRs. However, to address the regulatory requirements identified in Section 2.1 of this report, sufficient data on the safety features of the fuel is necessary to support licensing. This data may be obtained from test reactors (e.g., ATR and TREAT) provided that the tests and associated data are appropriate for assessing evaluation models (see Section 3.4 of this report).

As an alternative to the requirements of 10 CFR 50.43(e)(1), 10 CFR 50.43(e)(2) allows the use of a prototype plant to comply with testing requirements. To date, the NRC has not licensed a nuclear power plant as a prototype;~~,~~ however, it has been envisioned that this could be a pathway for licensing advanced reactors (see Appendix B to Enclosure 1 of the Regulatory Roadmap (NRC, 2017)). Use of a prototype plant may involve the application of additional NRC imposed requirements on siting, safety features, or operational conditions to protect the public and the plant staff from the possible consequences of accidents during the testing period.

NRC imposed requirements under 10 CFR 50.43(e)(2) may appear~~s~~ as license conditions. Any imposed conditions would consider risk-insights and any uncertainties associated with fuel

performance that are to be addressed by testing in the prototype plant~~reactor~~. Furthermore, the adequacy of these license conditions would be subject to the mandatory hearing (performed as part of plant licensing), and NRC staff expects license conditions to be an area of significant focus during a detailed technical review. Due to the significant impact that such a licensing strategy would have on the overall safety review of the plant~~facility~~, NRC staff encourages any applicant considering such a licensing strategy to have significant pre-application engagement on the topic of fuel qualification to ensure that there is a common understanding of associated technical, regulatory, and policy issues and to reduce regulatory uncertainty, as discussed in the NRC staff's draft Pre-application Engagement to Optimize Advanced Reactors Application Reviews white paper (NRC, 2021).

2.6 Assessment Framework

The top-down development of an assessment framework is not a novel approach in the regulatory process. Similar assessment frameworks have been developed in the code scaling, applicability, and uncertainty evaluation methodology (NRC, 1989), the evaluation model development and assessment process (NRC, 2005), and the "objectives hierarchy" discussed in NUREG/BR-0303, "Guidance for Performance-Based Regulation," issued December 2002 (NRC, 2002). Another top-down assessment framework, developed for thermal margin evaluations for LWRs, was based on many years of safety reviews (NRC, 2019). Assessment frameworks have facilitated safety reviews and have been shown to increase transparency about information needs, to promote efficiency by focusing attention on areas of recognized importance, and to clarify the logic behind decisions.

3 FUEL QUALIFICATION ASSESSMENT FRAMEWORK

This section on the fuel qualification assessment framework (FQAF) systematically identifies fuel safety criteria. The comprehensive list of safety criteria~~, called a fuel assessment framework,~~ is informed by existing regulatory requirements, regulatory guidance, and NRC staff experience with safety reviews for nuclear fuel in both LWRs and non-LWRs. The FQAF~~fuel assessment framework~~ is developed using a top-down approach that starts with the high-level goal (G) that the fuel ~~be~~is qualified ~~for use~~ and then decomposes this goal into subgoals. Meeting the subgoals indicates that the higher-level goal is met. Each subgoal can ~~either~~ be further decomposed into other subgoals, or if no further decomposition is deemed necessary, the subgoal may be considered a base goal and evidence must be provided to demonstrate that the base goal is satisfied. In this report, base goals are identified by the use of grey boxes.

As discussed in Section 1.4 of this report, the ~~assessment framework~~FQAF draws on regulatory experience gained from licensing solid fuel reactor designs (particularly LWR designs), results from advanced reactor fuel testing performed to-date, and AFQ considerations. An attempt has been made to develop a generically applicable set of base goals. However, some goals may not be applicable to all fuel types and reactor designs.
~~Consistent with the purpose of fuel qualification (see Section 1.1) and with a regulatory focus on safety, this report uses the following definition for fuel qualification:~~

> ~~Fuel is qualified for use if reasonable assurance exists that the fuel, fabricated in accordance with its specification, will perform as described in the safety analysis.~~

The definition of fuel qualification, from Section 1.2 of this report, ~~is statement~~ is captured figuratively in Figure 3-1, which decomposes fuel qualification into two supporting goals. These goals are further decomposed into lower level supporting goals, until criteria are obtained which can be directly verified by evidence. The subsections that follow describe the process, criteria, and associated evidence necessary to demonstrate fuel qualification.

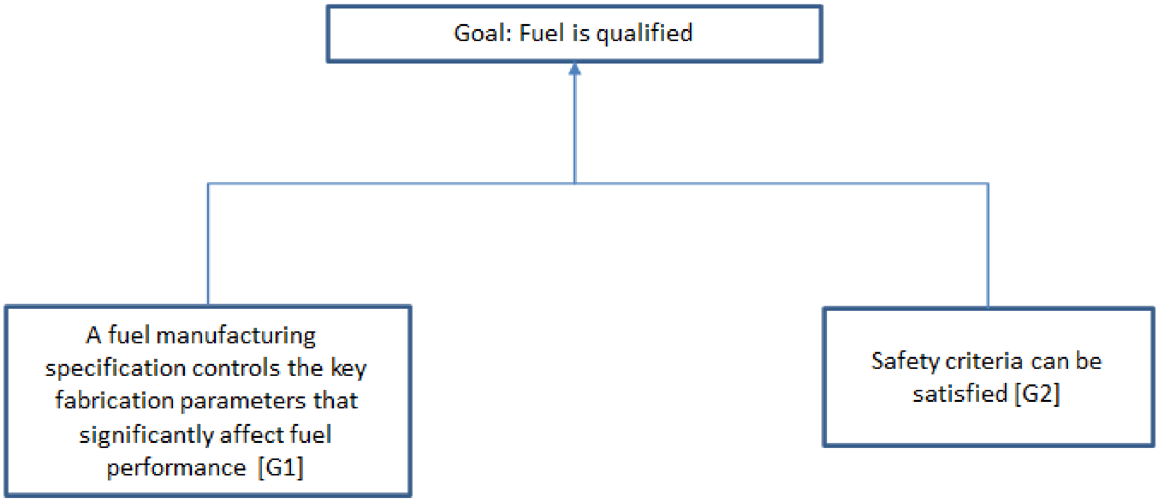

Figure 3-1 Decomposition of the Main Goal

3.1 G1—Fuel Manufacturing Specification

Fuel performance during normal operation and accident conditions can be highly sensitive to fuel manufacturing parameters. For example, failure criteria during reactivity-initiated accidents for LWRs with zirconium-based cladding depend upon the heat treatment of the cladding because of its impact on microstructure (NRC, 2020c). Similarly, key manufacturing parameters for TRISO particles have been identified that provide assurance of fuel performance during normal operation (EPRI, 2020). ~~Fuel performance during normal operation and accident conditions can be highly sensitive to the fuel fabrication process. For example, failure criteria during reactivity-initiated accidents for LWRs with zirconium-based cladding depend upon the heat treatment of the cladding because of its impact on microstructure (NRC, 2020c). Similarly, key manufacturing parameters have been identified for TRISO fuel that must be controlled to ensure satisfactory performance (EPRI, 2020).~~

The NRC ~~S~~staff recognizes that manufacturing processes for a nuclear fuel product may evolve over the product life cycle; therefore, a complete manufacturing specification is not expected as part of the licensing documentation. However, the licensing documentation should include sufficient information to ensure the control of key parameters affecting fuel performance during the manufacturing process. The goal G1 is decomposed as shown in Figure 3-2 to identify the specific types of information to be included in licensing documentation.

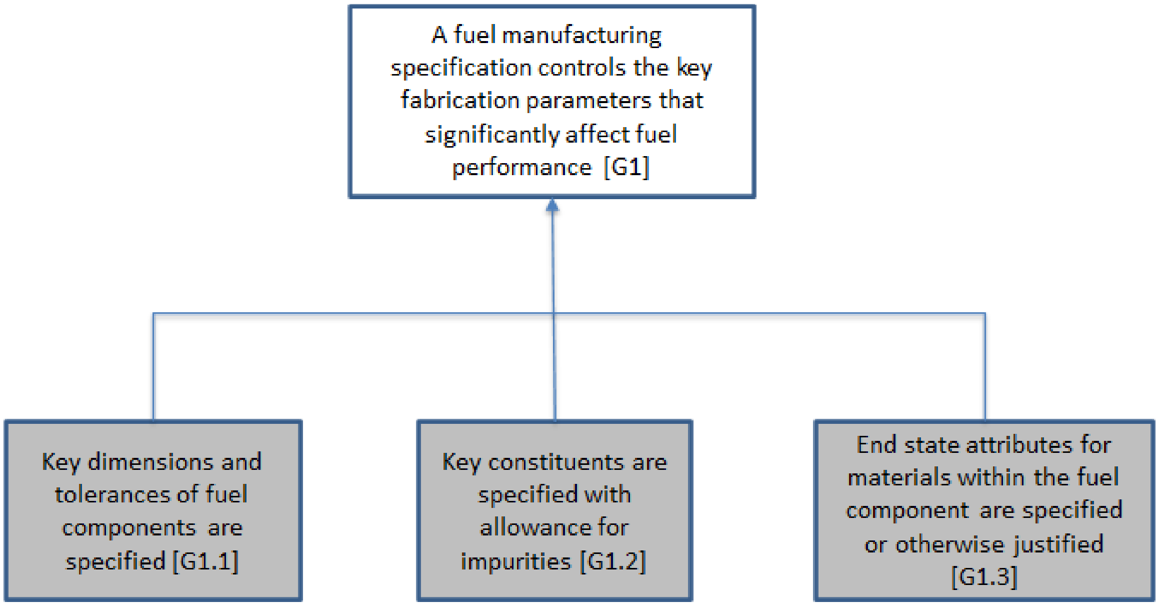

Figure 3-2 Decomposition of G1, "Fuel Manufacturing Specification"

3.1.1 G1.1–Dimensions

Key dimensions and tolerances for fuel components that affect performance should be specified. Consistent with the scope of this report, as discussed in Section 1.4 of this report~~3~~, these dimensions and tolerances should be specific to components that affect fuel life-limiting failure and degradation mechanisms that are due to irradiation and ~~irraditation~~irradiation assisted phenomena during reactor operation (e.g., fuel pellet and cladding dimensions, key assembly dimensions). It is recognized that some of dimensions can be controlled by an approved change process (e.g., General Electric Standard Application for Reactor Fuel (GESTAR) (GNF, 2019)~~)~~.

3.1.2 G1.2—Constituents

Key constituents of fuel components (e.g., uranium dioxide (UO_2) fuel, uranium-plutonium-zirconium fuel alloys with specified concentrations (U-Pu-10Zr), cladding material) should be specified, along with allowances for impurities.

3.1.3 G1.3—End State Attributes

End state attributes (e.g., microstructure) for the materials within fuel components ~~(e.g., microstructure)~~ should be specified or otherwise justified. The information necessary to capture the desired end state of the material may take several forms. For example, the identification and justification of key end-state parameters has been an acceptable means for providing assurance for TRISO particles for performance under conditions of normal operation (EPRI, 2020). Alternatively, specific manufacturing processes (e.g., cold-working, heat treatments, acid pickling, deposition techniques) that are essential to create the microstructure may be indicated in lieu of end state attributes. ~~For example, specific manufacturing processes (e.g., cold-working, heat treatments, acid pickling, deposition techniques) that are essential to create the microstructure may be indicated in lieu of end state attributes.~~ In some cases, it may be preferable to use performance-based end state attributes that can be supported through periodic testing and reporting (NRC, 2016). Additionally, it may be possible to demonstrate

insensitivity to manufacturing processes so that end state attributes need not be specified in licensing documentation. Licensing documentation should provide sufficient justification for cases where a specific material is insensitive to manufacturing processes.

3.2 G2—Safety Criteria

An evaluation of the ~~safety case~~ fuel involves an assessment against safety criteria, which are generally associated with protection against the release of radioactive material but also address the fundamental safety functions of heat removal and reactivity control. In general, many safety criteria for nuclear fuel depend on the events to which the fuel is subjected. Specifically, nuclear fuel is expected to retain its integrity under conditions of normal operation, including the effects of AOOs, but some degree of fuel failure can be accommodated for low-frequency design-basis accident conditions (i.e., those not expected to occur during the life of the plant). The goal G2 is decomposed as shown in Figure 3-3 to address the varying types of safety criteria for the range of events for which nuclear fuel should be qualified.

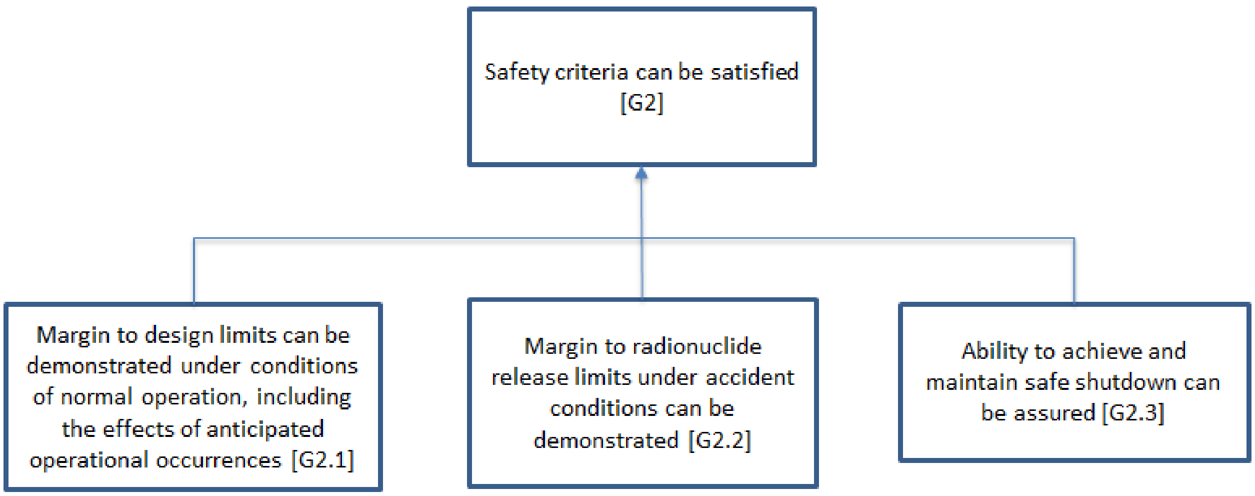

Figure 3-3 Decomposition of G2, "Safety Criteria"

3.2.1 G2.1—Design Limits during Normal Operation and Anticipated Operational Occurrences

Fuel integrity is expected to remain intact under conditions of normal operation, including the effects of AOOs. Alternatively, some designs may use SARRDLs, which allow a small degree of radionuclide release from the fuel (NRC, 2018b). Multiple degradation mechanisms and failure modes may exist; limits need to be established to protect against all of them. At the highest level, the assessment of a fuel against design limits for normal operation and AOOs requires knowledge of the conditions that the fuel is exposed to (i.e., the performance envelope) and a method to assess the fuel performance under those conditions (i.e., an evaluation model). These supporting goals, shown in Figure 3-4, are discussed below.

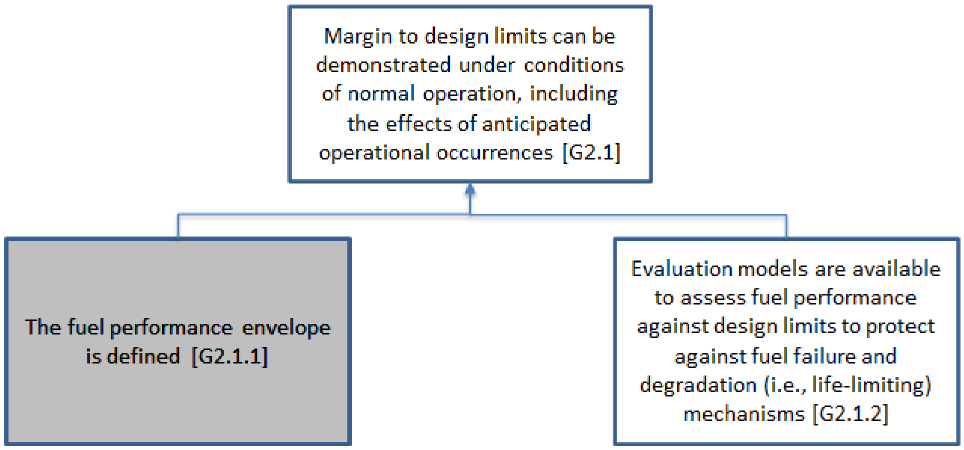

Figure 3-4 Decomposition of G2.1, "Design Limits During Normal Operation and AOOs"

3.2.1.1 G2.1.1—Definition of Fuel Performance Envelope

The fuel performance envelope specifies the environmental conditions and radiation exposure under which the fuel is required to perform. This performance envelope informs the safety analysis, ~~and~~ technical specifications and/or operating limits for the design. ~~(i.e., limiting conditions for operation)~~[9] It is noted that irradiation-induced growth and fission product swelling of fuel components are often life-limiting phenomena for the fuel design. The envelope may be specified by fuel designers and may constrain the design of the reactor and associated systems. Alternatively, a reactor design may be proposed that imposes constraints on fuel performance. In support of G2.1, the goal G2.1.1 can be met by specifying the conditions (e.g., temperatures, pressures, power), exposure, and transient conditions that the fuel is expected to encounter during normal operation, including AOOs. Additionally, G2.1.1 supports G2.2, which addresses the fuel contribution to the source term during design-basis accidents, as discussed in Section 3.2.2.1 of this report. Accordingly, this goal can be fully satisfied by specifying the conditions the fuel is expected to encounter during normal operation, AOOs, and design-basis accidents.

3.2.1.2 G2.1.2—Evaluation Model

This goal—that evaluation models are available to assess fuel performance against design limits to protect against fuel failure and degradation mechanisms—requires the specification of means of evaluating fuel for performance, failure, and degradation. The assessment of evaluation models supports several goals and is further decomposed. Therefore, Section 3.3 of this report provides a separate assessment framework for evaluation models. G2.1.2 is satisfied by meeting the supporting goals in that framework for fuel performance during normal operation and AOOs.

[9] 10 CFR 50.36(c)(2)(ii(B), Criterion 2 requires a limiting condition for operation for a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3.2.2 G2.2—Radionuclide Release Limits

Radiological consequences under postulated accident conditions are an essential consideration in nuclear ~~power plant~~reactor licensing. Under postulated accident conditions, some fuel failure is possible, which contributes to the accident source term. As radionuclide inventory originates from the nuclear fuel, fuel qualification should include characterizing the behavior of the fuel under accident conditions, so that its contribution to the accident source term can be determined in a suitably conservative manner. Accordingly, the goal G2.2—the ability to demonstrate margin to radionuclide release limits under accident conditions, in relation to fuel qualification—is supported by three goals related to the fuel contribution to the accident source term. These goals, shown in Figure 3-5 (along with G2.1.1, which also supports G2.2), are discussed further below.

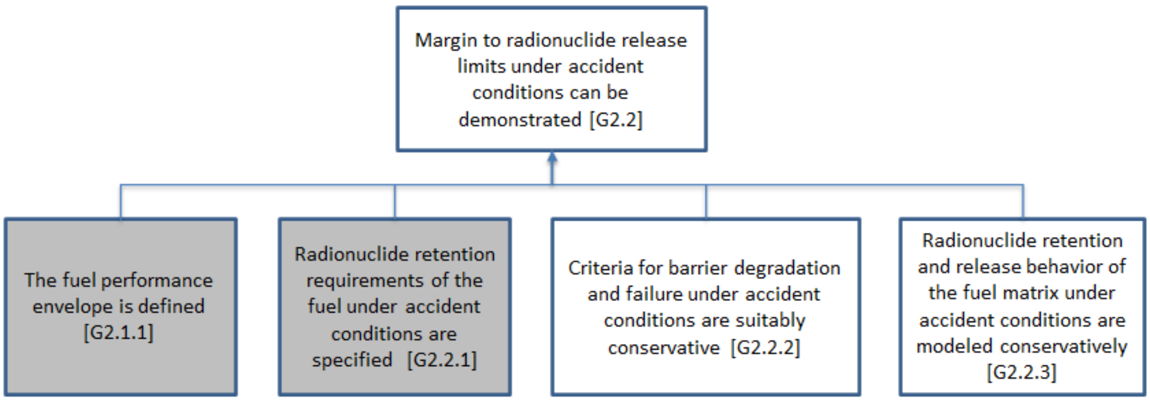

Figure 3-5 Decomposition of G2.2, "Radionuclide Release Limits"

3.2.2.1 G2.1.1—Definition of Fuel Performance Envelope

Section 3.2.1.1 of this report already discussed G2.1.1. In support of G2.2, this goal can be satisfied by specifying the design-basis accident conditions to which the fuel is subjected. Design-basis accident conditions depend on reactor design; however, as discussed in Section 3.2.1.1 of this report, conditions to which the fuel is subjected during design-basis accidents may be specified independent of the reactor design, leading to constraints on the design of the reactor and associated systems. For example, bounding conditions, (e.g., temperatures, pressures, transient power) may be provided that would encompass the performance of an unspecified reactor design. The types of design-basis accident conditions that should be considered include transient overpower events (e.g., reactivity-initiated accidents), transient undercooling events (e.g., loss-of-coolant accidents), and externally applied loads (e.g., fuel handling, transportation, seismic activity, and major piping failures).

3.2.2.2 G2.2.1—Radionuclide Retention Requirements

The role of nuclear fuel in the protection against the release of radioactivity ~~safety case~~ can vary between reactor designs and fuel types. For example, traditional LWR fuel that uses UO_2 pellets with zircalloy cladding is not credited to retain cladding integrity under large-break loss-of-coolant accidents[10], while advanced reactor designs may credit retention of radionuclides within

[10] NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," (NRC, 1995) states that, "Assuming that the coolant loss cannot be accommodated by the reactor coolant makeup systems or the emergency core cooling systems, fuel cladding failure would occur with the release of radioactivity located in the gap between the fuel pellet and the fuel cladding."

the fuel under accident conditions. Additionally, plant site characteristics such as proximity to population and weather patterns may further influence radionuclide retention requirements (even for the same reactor and fuel design). To satisfy G2.2.1, the ~~degree of~~ radionuclide retention requirements of ~~within~~ the fuel system under accident conditions should be specified.

3.2.2.3 G2.2.2—Criteria for Barrier Degradation

Radionuclide barrier (e.g. fuel cladding) failure and degradation mechanisms under accident conditions (e.g., pellet-clad mechanical interaction (PCMI) and high enthalpy failure, temperature-induced reactions and phase transformations) must be understood when the design credits retention of barrier integrity (e.g., during reactivity-initiated accidents in LWRs, or considering the potential for fission product attack of the silicon carbide layer in TRISO fuel at high temperatures). As such, the goal G2.2.2 is decomposed into two supporting goals, shown in Figure 3-6.

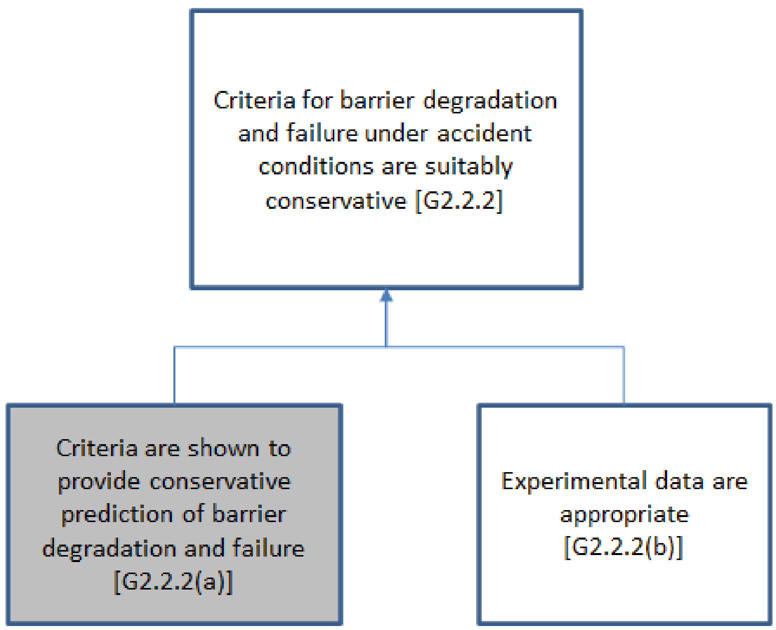

Figure 3-6 Decomposition of G2.2.2, "Criteria for Barrier Degradation"

3.2.2.3.1 G2.2.2(a)—Conservative Criteria

Criteria used to determine barrier degradation should be suitably conservative. These criteria are expected to be established based on transient testing and irradiated fuel samples, as discussed under G2.2.2(b). Ideally, sufficient data would be available to demonstrate margin with uncertainty at a statistical confidence level using an independent set of data~~Ideally, to establish a statistical confidence level (e.g., 95/95), criteria would be established through a regression analysis using experimental data, then validated by assessment against a separate and independent set of data~~ (see Section 3.4.1 (Experimental Data (ED) G1) of this report for a discussion on data independence). However, this ideal scenario may not be realized due to challenges associated with obtaining irradiated fuel samples and conducting transient testing for design-basis accident conditions. The amount of experimental data supporting the criteria should be proportional to the degree of understanding of key degradation and performance phenomena (NRC, 2020c). If the data collected are not sufficient to support statistical modeling, a conservative or bounding approach may be required.

3.2.2.3.2 G2.2.2(b)—Experimental Data

This goal is satisfied through an evaluation against the experimental data assessment framework in Section 3.4 of this report.

3.2.2.4 G2.2.3—Conservative Modeling of Radionuclide Retention and Release

Consistent with the requirements specified as part of G2.2.1 and discussed in Section 3.2.2.2 of this report, radionuclide retention and release behavior of the fuel under accident conditions should be modeled conservatively. This goal is related to the barrier degradation criteria specified in G2.2.2 and discussed in Section 3.2.2.3 of this report, but it differs in its focus on radionuclide retention within the fuel matrix (e.g., UO_2 pellet or uranium alloy with 10 percent zirconium (U-10Zr) fuel ingot) or fuel particle (e.g., fuel compact for a TRISO-based fuel). This goal is decomposed into two supporting goals, as shown in Figure 3-7.

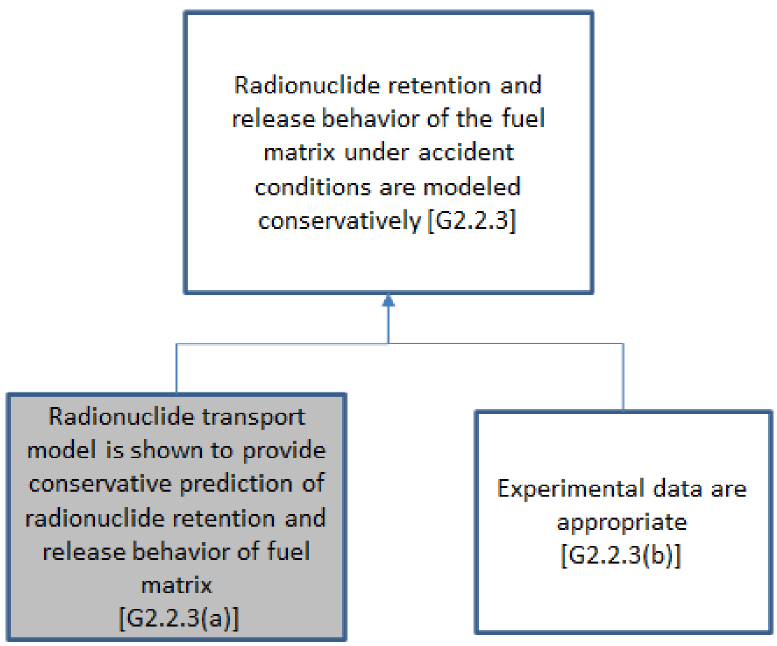

Figure 3-7 Decomposition of G2.2.3, "Conservative Modeling of Radionuclide Retention and Release"

3.2.2.4.1 G2.2.3(a)—Conservative Transport Model

The model of radionuclide transport within the fuel matrix should be conservative. As in the case of barrier degradation criteria, discussed in Section 3.2.2.3.1 of this report, challenges associated with obtaining and testing irradiated fuel samples may make it difficult to obtain sufficient data to characterize the transport model in a statistical manner; therefore, conservative or bounding estimates may be required. Additionally, previous source term models for LWRs have generally included some degree of expert judgment. A clarifying example of how to develop a suitably conservative radionuclide transport model is available in regulatory guidance on accident source terms (NRC, 2000).

3.2.2.4.2 G2.2.3(b)—Experimental Data

This goal is satisfied through an evaluation against the experimental data assessment framework in Section 3.4 of this report.

3.2.3 G2.3—Safe Shutdown

Safe shutdown of a nuclear plant refers to a state in which the reactor is subcritical, decay heat is being removed, and radionuclide inventory is contained. The international atomic energy agency (IAEA) refers to this as a *safe state* (IAEA, 2018). The ability to achieve safe shutdown in any scenario needs to be assured. Therefore, criteria need to be established to ensure that a coolable geometry is maintained in all scenarios and that fuel system damage is never so severe as to preclude the insertion of negative reactivity sufficient to hold the reactor subcritical~~prevent control element (e.g., control rod) insertion when required~~. These supporting goals, captured in Figure 3-8, are discussed below.

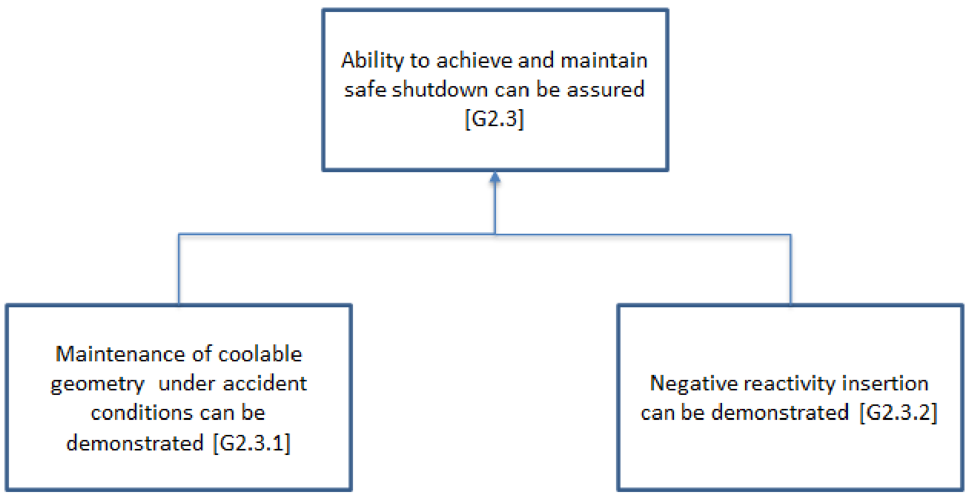

Figure 3-8 Decomposition of G2.3, "Safe Shutdown"

3.2.3.1 G2.3.1—Maintaining Coolable Geometry

The maintenance of a coolable geometry is identified as a supporting goal in achieving and maintaining safe shutdown. It is further decomposed into the supporting goals shown in Figure 3-9, which are discussed below.

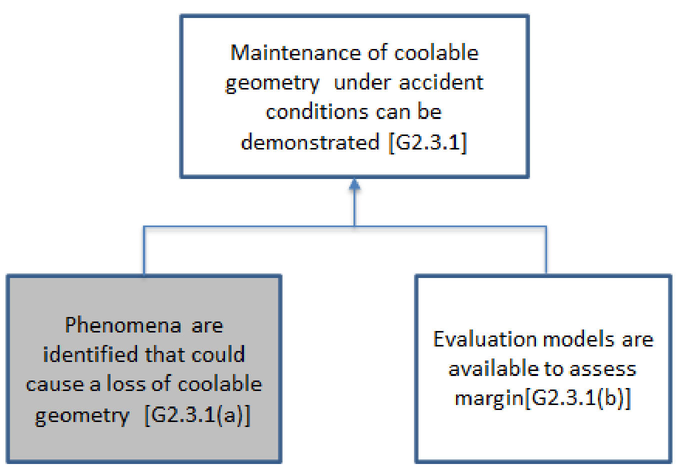

Figure 3-9 Decomposition of G2.3.1, "Maintaining Coolable Geometry"

3.2.3.1.1 G2.3.1(a)—Identification of Phenomena

Phenomena that could cause the loss of coolable geometry should be specified. ~~Existing~~ NRC regulations and guidance applicable to design basis accidents specify some acceptance criteria for these events that are intended to prevent such phenomena from significantly altering core geometry under postulated accident conditions (NRC, 2007). Examples of phenomena that could cause the loss of coolable geometry include: (1) fuel melt, (2) fuel swelling and fuel pellet and cladding fragmentation and dispersal during transient overpower events, and (3) loss of cladding ductility or long-term cladding phase stability during loss-of-coolant accidents.

3.2.3.1.2 G2.3.1(b)—Evaluation Models

Several evaluation models may be needed to demonstrate that coolable geometry is maintained. These models typically involve the use of conservative criteria and the evidence needed to meet this goal depends on the associated phenomena. For example, a conservatively chosen criterion such as the onset of fuel melting should not require a detailed evaluation model supported by integral testing, but an empirically based criterion such as energy deposition for fuel dispersal or peak cladding temperature for cladding embrittlement requires the demonstration of an appropriate margin against experimental data. Historical examples of acceptable empirical criteria include those developed for transient overpower (NRC, 2020c) and loss-of-coolant accidents (Hache & Chung, 2000). In addition to these empirical models for demonstrating a coolable geometry, analytical models have been used to demonstrate that coolable geometry is maintained for internal and external events (Framatome, 2018).

The evaluations performed to demonstrate coolable geometry vary in terms of complexity, f~~o~~rom simple conservative criteria to detailed dynamic response models. The most general case that applies to all these situations is the generic evaluation model assessment discussed in Section 3.3 of this report. Accordingly, this goal is satisfied through a comparative assessment against the evaluation model assessment framework in Section 3.3 of this report. The application of the evaluation model assessment framework should follow a graded approach in accordance with the level of understanding of the physical phenomena and conservatism in the criteria.

3.2.3.2 G2.3.2—Negative Reactivity~~Control Element~~ Insertion

~~Control element~~Negative reactivity insertion is identified as a supporting goal in achieving and maintaining safe shutdown. It is further decomposed into the supporting goals shown in Figure 3-10, which are discussed below.

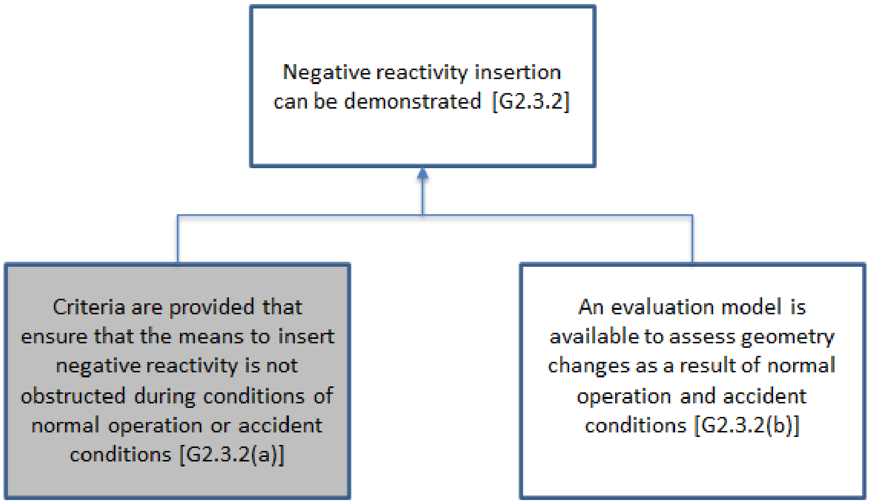

Figure 3-10 Decomposition of G2.3.2, "Negative Reactivity~~Control Element~~ Insertion"

3.2.3.2.1 G2.3.2(a)—Identification of Criteria

Criteria should be specified to ensure that the means to insert negative reactivity, sufficie~~i~~nt to hold the reactor subcritical under long term core cooling conditions, ~~control element insertion path~~ is not obstructed during normal operation or accident conditions. These criteria should ~~consider~~be graded in accordance with the degree to which the fuel can obstruct the insertion of negative reactivity. Additionally, these criteria should consider loads from both internal and external (e.g., seismic) events when applicable. An example of such a criterion for traditional LWRs is the stress limit imposed on the control rod guide tubes to inhibit distortion of the insertion path.

3.2.3.2.2 G2.3.2(b)—Evaluation Model

The evaluation performed to demonstrate that negative reactivity~~control element~~ insertion can be assured has typically involved a stress analysis to ensure that the negative reactivity~~control element~~ insertion path is not deformed as a result of internal and external events. This is typically done using a separate evaluation model. Accordingly, this goal is satisfied through a comparative assessment against the evaluation model assessment framework in Section 3.3 of this report.

3.3 Assessment Framework for Evaluation Models

The term "evaluation model" here is used in the generic sense. Typically, an evaluation model is an analytical tool, a computer code, or a combination of such tools. However, the use of a sophisticated tool such as a computer code may not be necessary to evaluate fuel performance. For example, a simple mathematical expression or set of data can serve as an evaluation model, if sufficient evidence exists to support its use.

The evaluation model assessment framework developed here is designed to be generically applicable. In particular, it supports G2.1.2, which addresses the evaluation of design limits under conditions of normal operation and AOOs, G2.3.1(b), which addresses maintaining coolable geometry, and G2.3.2(b), which addresses negative reactivity~~control element~~ insertion. The evaluation model assessment framework presented here overlaps conceptually with the goals previously established for criteria for barrier degradation (Section 3.2.2.3 of this report)

and radionuclide retention and release (Section 3.2.2.4 of this report). The latter two goals, however, have historically involved empirical evaluation models based on destructive testing using irradiated nuclear fuel under accident conditions. Accordingly, goals for barrier degradation and radionuclide retention and release are provided separately from the evaluation model assessment framework of this section.

The top-level goal of an acceptable evaluation model is supported by the goals of (1) adequate modeling capabilities and (2) assessment against experimental data. These supporting goals are shown in Figure 3-11 and discussed below.

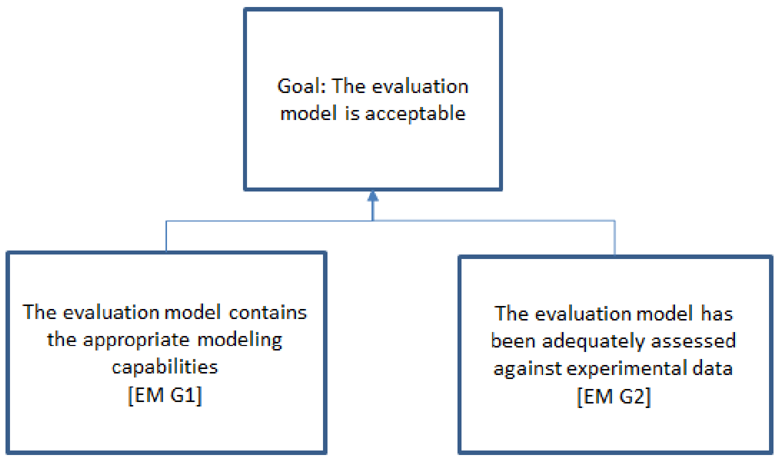

Figure 3-11 Decomposition of the Main Goal for Evaluation Model Assessment

3.3.1 EM G1—Evaluation Model Capabilities

The evaluation model capabilities goal is decomposed into three supporting goals as shown in Figure 3-12. This decomposition is informed by the predictive capability maturity model (PCMM) framework, which identifies "representation and geometric fidelity" and "physics and material model fidelity" as assessment elements (SAND, 2007). The evaluation model assessment framework also considers other elements of the PCMM framework. Specifically, EM G2 addresses "model validation" and "uncertainty quantification and sensitivity analysis"; see Section 3.3.2 of this report. The remaining elements of the PCMM framework, "code verification" and "solution verification," are expected to be addressed as part of a quality assurance program for the design, analysis, and fabrication of a nuclear power facility. The goals supporting EM G1, shown in Figure 3-12, are discussed below.

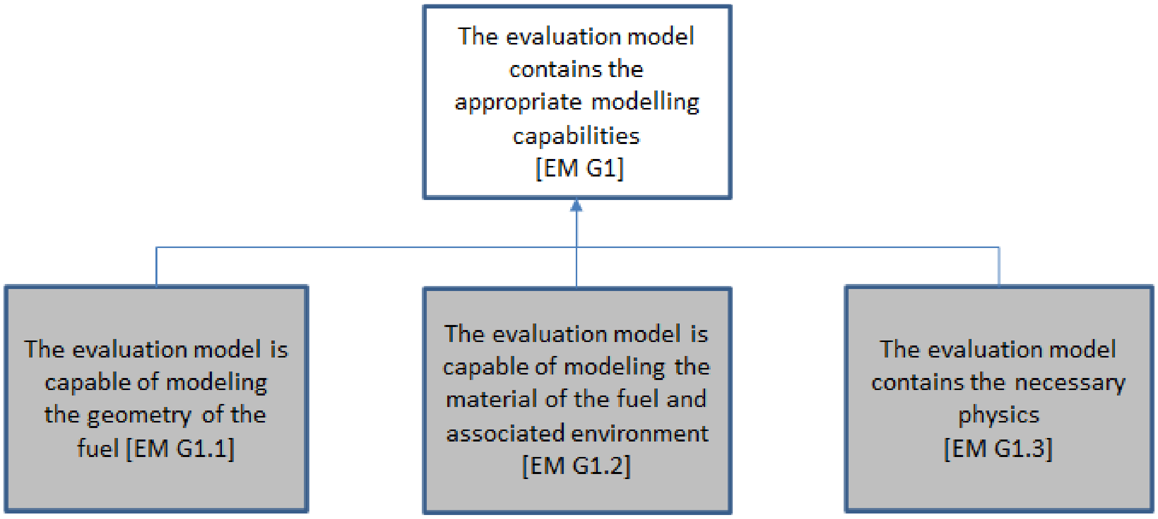

Figure 3-12 Decomposition of EM G1, "Evaluation Model Capabilities"

3.3.1.1 EM G1.1—Geometry Modeling

The evaluation model should be capable of modeling the geometry of the fuel system. Table 3 of the PCMM provides guidance on the levels of maturity needed to assess the geometry, including consideration of peer review (SAND, 2007). It is recognized that some fuel designs may require simplifying assumptions to address difficulties in geometric modeling. For example, TRISO-based particulate fuel involves coupled phenomena occurring at different geometric scales (e.g., micro-scale within the TRISO particle, meso-scale within the fuel compact, and macro-scale within the reactor core). Geometric modeling for such particulate fuel could involve simplifications and assumptions that a less heterogeneous fuel design may not require. Additionally, the evaluation model should be able to capture geometric changes due to irradiation and exposure to the in-reactor environment (e.g., fuel swelling, cladding creep, oxide layer growth). Irrespective of imposed simplifications, the geometric modeling scheme should be appropriately justified, and the integrated evaluation model should be validated through the assessment process under EM G2.

3.3.1.2 EM G1.2—Material Modeling

The evaluation model should be capable of modeling material properties of the fuel system and its surrounding environment. This includes changes in material properties due to irradiation and exposure to the in-reactor environment (e.g., thermal conductivity degradation in nuclear fuel, changes to melting temperature, eutectic formation, changes to Young's modulus). Table 3 of the PCMM provides guidance on the levels of maturity needed to assess the material modeling, including considerations for model calibration against test data and peer review (SAND, 2007). The material modeling scheme should be justified, and the integrated evaluation model should be validated through the assessment process under EM G2.

3.3.1.3 EM G1.3—Physics Modeling

The evaluation model should be capable of modeling the physical processes that affect fuel performance. This goal requires knowledge of failure mechanisms, including changes due to irradiation and exposure to the in-reactor environment for the specified fuel, as well as fuel contribution to the SARRDL, if applicable. The evaluation model is expected to include sufficient physics modeling to address known degradation mechanisms (e.g., cladding oxidation and

hydrogen pickup, fuel rod internal pressure, cladding strain, fuel assembly growth and wear, stress and fatigue for fuel components). Table 3 of the PCMM provides guidance on the levels of maturity needed to assess the physics modeling, including considerations for model calibration against test data and peer review (SAND, 2007). The physics models incorporated into the evaluation model should be justified, and the integrated evaluation model should be validated through the assessment process under EM G2. Means of justification include the use of an expert panel to develop a PIRT (PNNL, 2019) and internal review based on past experience, legacy data (ANL, 2018), or separate-effects testing (Beausoleil II, Povirk, & Curnutt, 2020) (Petrie, Burns, Raftery, Nelson, & Terrani, 2019).

3.3.2 EM G2—Evaluation Model Assessment

Evaluation model assessment is an essential process that provides confidence in the application of the evaluation model. To ensure that evaluation model predictions are suitably conservative, they should be assessed against appropriate experimental data. For statistically based modeling approaches, any bias or uncertainty in the evaluation model prediction should be adequately quantified, so that design and safety analyses can account for such bias or uncertainty. For conservative modeling approaches, the evaluation model should suitably bound the experimental data. The assessment process comprises two supporting goals, shown in Figure 3-13, which are discussed below.

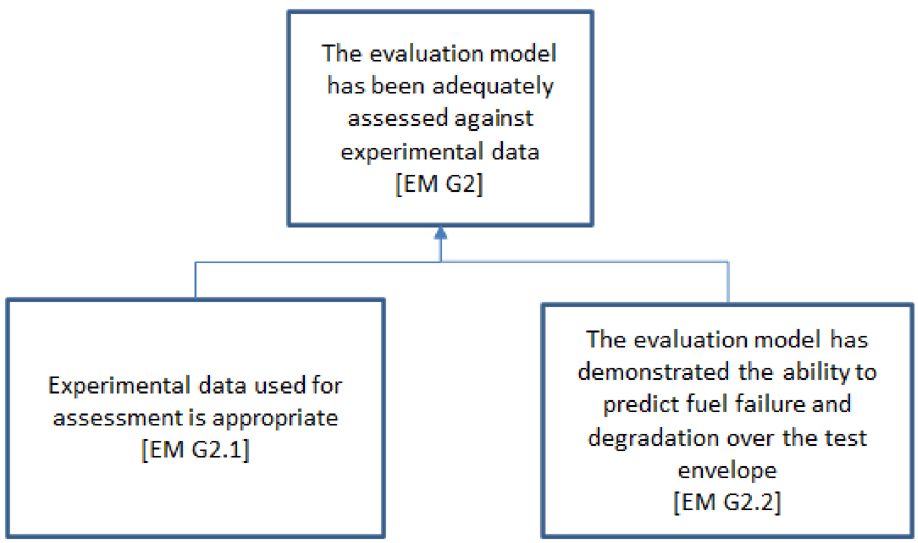

Figure 3-13 Decomposition of EM G2, "Evaluation Model Assessment"

3.3.2.1 EM G2.1—Experimental Data

This goal is satisfied through an evaluation against the experimental data assessment framework in Section 3.4 of this report.

3.3.2.2 EM G2.2—Demonstrated Prediction Ability over Test Envelope

EM G2.2 involves the comparison of evaluation model predictions against experimental data, which should establish uncertainties and biases and identify limitations in the applicability of the evaluation model. EM G2.2 is satisfied by meeting the four supporting goals shown in Figure 3-14, which are discussed below.

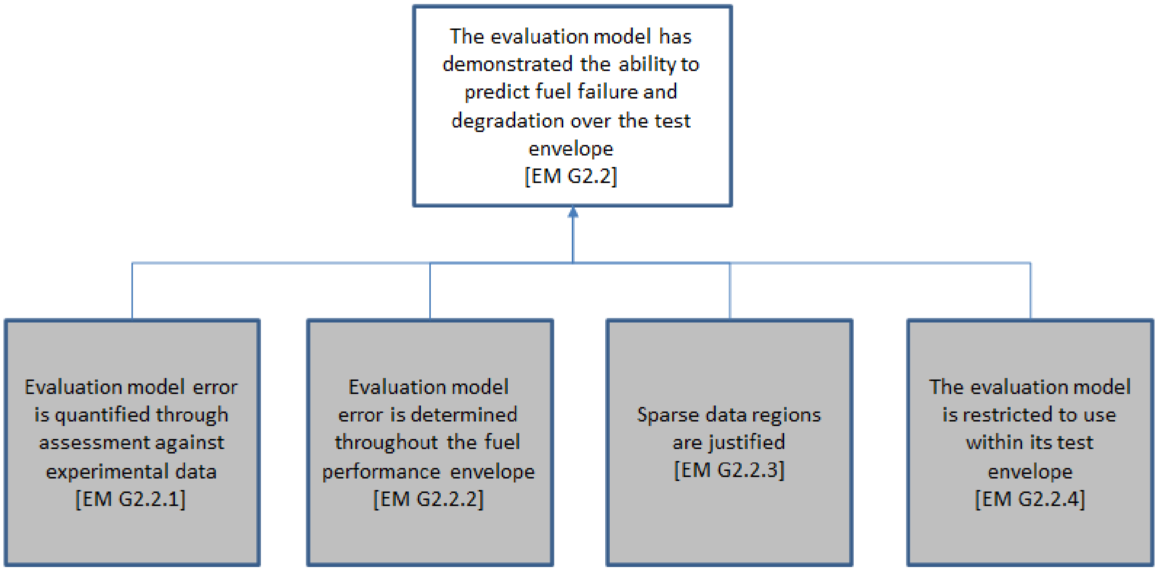

Figure 3-14 Decomposition of EM G2.2, "Demonstrated Prediction Ability Over Test Envelope"

3.3.2.2.1 EM G2.2.1—Quantification of Error

Uncertainties and biases for figures of merit need to be sufficiently understood to establish confidence in the evaluation model. It is expected that, to determine uncertainties and biases, the predictions of the evaluation model for assessment cases will be compared against assessment data, and the differences between measured and predicted values will be quantified. If sufficient data exist, then statistical confidence levels can be placed on the uncertainties of the evaluation model predictions. However, a more bounding or conservative approach can also be taken (e.g., applying a bias or penalty to the model predictions, showing that the model is inherently conservative). EM G2.2.1 can be satisfied by a statement on the evaluation model biases and uncertainties, along with justification through a quantification of the ratio of predicted to measured values for assessment cases.

3.3.2.2.2 EM G2.2.2—Span of Validation Data

Assessment data should be distributed throughout the fuel performance envelope. The fuel performance envelope, discussed in Sections 3.2.1.1 and 3.2.2.1 of this report, is used to specify the test envelope; accordingly, assessment data should be available to assess the evaluation model over the entire span of the performance envelope. However, it is recognized that certain regions of the fuel performance envelope may not require data. For example, post-irradiation examination of an integral test specimen may not be necessary for low-burnup fuel. In such cases, it may suffice to provide justification that those regions do not require data (e.g., that limiting phenomena are known not to be present below a specified burnup). EM G2.2.2 can be satisfied by demonstrating that assessment data are available over the entire performance envelope, and by justifying any gaps in assessment data.

3.3.2.2.3 EM G2.2.3—Data Density

Assessment data should be appropriately distributed throughout the fuel performance envelope. As discussed in Section 3.3.2.2.2 of this report, it may be acceptable to have regions in the

performance envelope where the evaluation model is not directly supported by assessment data from integral experiments. However, in regions that do require assessment data, a sufficient number of data points should be available for assessment of the evaluation model. It is reasonable to expect data density to be greater near conditions of normal operation, as fuel designers may require additional data to satisfy fuel reliability targets. However, any sparse data regions (i.e., regions of low data density) in the fuel performance envelope need adequate justification. EM G.2.2.3 can be satisfied by justifying the data density throughout the fuel performance window.

3.3.2.2.4 EM G2.2.4—Restricted Domain

Use of the evaluation model should be restricted to application domains for which the model has been assessed. Application of an evaluation model outside of the supporting test envelope (see Section 3.4.2 of this report) may be justified based on physical arguments (e.g., that the evaluation model provides a simplified or bounding treatment of physical phenomena). Justification for extrapolation of a model outside of the test envelope is strengthened by the use of physics-based models, such as those discussed in Section 2.3 of this report, which are informed by fundamental information about fuel evolution and behavior, as opposed to empirically derived models (Terrani, et al., 2020). EM G2.2.4 can be satisfied by specifying the application domain of the evaluation model as supported by the test envelope and by additional physical arguments as necessary.

3.4 Assessment Framework for Experimental Data

The assessment of experimental data is the largest area of review for fuel qualification. The assessment framework developed here supports all goals requiring evaluations against assessment data. Because a fuel qualification program involves several types of experiments (e.g., steady-state irradiation of integral test specimens, transient ramp testing, design-basis accident testing), and because of transient test facility (e.g., ATR and TREAT) limitations and challenges associated with irradiated fuel testing, it is recognized that the level of evidence expected to support a goal can vary depending on the type of data collected. The assessment framework presented in this section discusses this variance in the level of evidence as applicable. The main goal for assessment data is decomposed, as shown in Figure 3-15, into four supporting goals, which are discussed below.

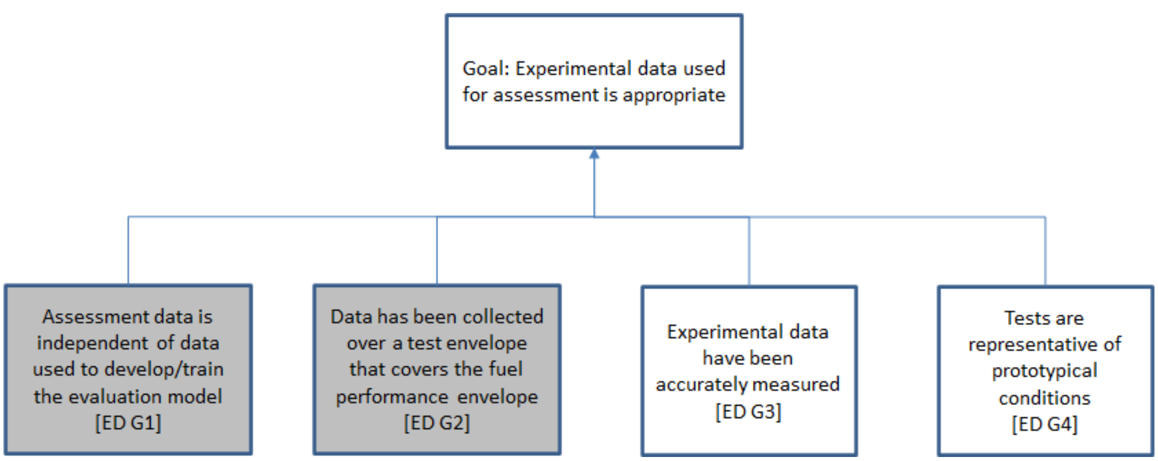

Figure 3-15 Decomposition of the Main Goal for Data Assessment

3.4.1 ED G1—Independence of Validation Data

Assessment data consist of experimentally measured values that are used to quantify the error in the evaluation model. Ideally, assessment data should be independent from any data used in the development (i.e., training) of the evaluation model. Although it may seem appropriate to use training data, training data cannot provide an accurate assessment because the evaluation model has already been "tuned" to those data. That is, quantifying the error of the training data would only show how well the model can predict the data used to generate it, not how well the model can predict data not used to generate it. Substantially more data points appear in the application domain (an infinite number) than were used to generate the model, and these are the points of most interest in future uses of the model; therefore, the focus should be on estimating the error over those points, not on the points used to generate the model. Thus, experimental data that were not used to train the model should be held in reserve and used to validate the model. Maintaining validation data separate from the model development process helps avoid a potential source of bias that could provide a distorted indication of the model's accuracy for future uses.

In some instances, however, the validation data and the training data are one and the same. Methods exist ~~in machine learning~~ for determining whether the selection of the training data affects the resulting uncertainty; such methods include random subsampling and k-fold cross-validation. In each of these methods, the available data are randomly separated into subsets of training and validation data. The training data are used to develop the coefficients of the model, and the validation data are used to determine the overall uncertainty of the model. The process is then repeated with different randomly selected training and validation data sets. These methods can provide reasonable estimates of the impact of using the same data for training and validation.

The discussion of data independence has so far considered scenarios where a sufficient number of data points exist to train and validate a model using statistical approaches (i.e., model regression and the calculation of confidence intervals). It is recognized, however, that only limited data may be available because of the challenges associated with obtaining irradiated fuel samples. Experience from transient overpower testing has shown that it may be acceptable to develop criteria without separating the data into training and validation sets (NRC, 2020c). Similarly, fission gas release and swelling models have been proposed based on a limited amount of test data (Lee, Kim, & Jung, 2001). ED G1 can be satisfied by demonstrating that the data used in the evaluation model assessment are sufficiently independent.

3.4.2 ED G2—Test Envelope

Data should be collected over a test envelope that spans the performance envelope (see Section 3.2.1.1 of this report). The performance envelope should address normal operation, AOOs, and postulated accident conditions. The development of the test envelope should consider (1) steady-state integral testing of the fuel system in a prototypical environment, (2) high-power and undercooling tests to address AOO conditions and to assess design margins, (3) power ramp testing to assess fuel performance during anticipated power changes, and (4) design-basis accident tests to establish margin to fuel breach and contribution to the source term under accident conditions. Typical design-basis accident scenarios of interest include overpower events (e.g., reactivity-initiated accidents) and undercooling events (e.g., loss-of-coolant accidents).

Many of the data necessary for fuel qualification come from irradiated test specimens. However, test specimens at the desired conditions may sometimes be unavailable. In such situations, it may be possible to use lead test specimens to extend the burnup limits of a fuel type. In some cases, direct examination of lead test specimens may provide a basis to support extending applicability of an evaluation model to a new burnup range. In other cases, irradiated lead test specimens may become the subject of subsequent tests under transient or accident conditions to assess evaluation models applicable under such conditions.

Lead test specimen programs have traditionally allowed for the placement of a limited number of test specimens in nonlimiting regions of the reactor core to maximize the safety margin. However, an extended use of lead test specimens (e.g., relaxation of the number and/or location of the test specimens) may be allowable if justified by a safety analysis that includes margin to account for the uncertainty in the performance of fuel above its burnup limit. The use of fuel above its qualified limit should be supported by sufficient monitoring to detect potential failures. Methods are available, such as gas tagging (McCormick & Schenter, 1974) (Pollack, Lewis, & Kelly, 2013), that can be used to identify the precise source of potential fuel failures. Additionally, if lead test specimens are subjected to conditions beyond existing data ranges, a licensing review may be necessary to ensure the appropriate level of safety before the extended limits are applied to the fuel design. ED G2 can be satisfied by demonstrating that the test envelope addresses the necessary performance envelope for the fuel design.

3.4.3 ED G3—Data Measurement

An understanding of measurement accuracy is essential to establish confidence in the data used to develop and assess evaluation models. This goal is decomposed, as shown in Figure 3-16, into three supporting goals, which are discussed below.

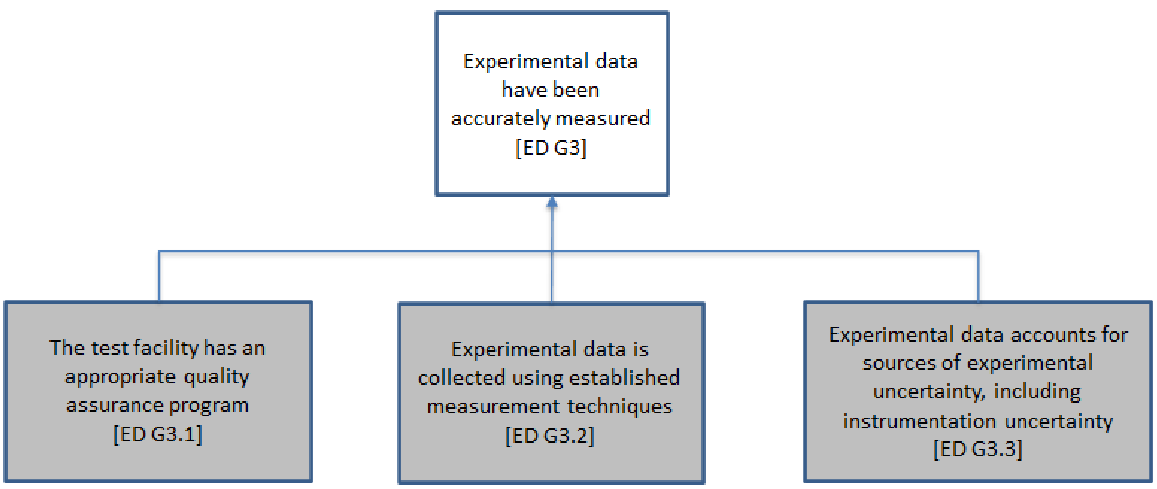

Figure 3-16 Decomposition of ED G3, "Data Measurement"

3.4.3.1 ED G3.1—Test Facility Quality Assurance

Experimental data should be collected under an appropriate quality assurance program that meets applicable regulatory requirements. Standards such as the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA)-1 are available for test facility quality assurance. NQA-1 also provides criteria for assessing historical data to facilitate compliance with quality assurance requirements (ANL, 2020). Alternative approaches, such as commercial grade dedication, may also be an acceptable means for justifying that data was

collected under an appropriate quality assurance program. ED G3.1 can be satisfied by demonstrating that data were collected under an appropriate quality assurance program or by otherwise justifying the use of existing data. ~~Experimental data should be collected under an appropriate quality assurance program. Standards such as the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA)-1 are available for test facility quality assurance. Provisions may also be applied to existing data to make them compliant with quality assurance requirements (ANL, 2020). ED G3.1 can be satisfied by demonstrating that data were collected under an appropriate quality assurance program or by otherwise justifying the use of existing data.~~

3.4.3.2 ED G3.2—Measurement Techniques

Data should be collected using established or otherwise proven measurement techniques. The use of novel or first-of-a-kind measurement techniques should be adequately justified. ED G3.2 can be satisfied by specifying the measurement techniques and justifying the use of any novel or first-of-a-kind techniques.

3.4.3.3 ED G3.3—Experimental Uncertainties

An error analysis should be performed to assess sources of bias and uncertainty in each experiment. Measurement uncertainty should be quantified when possible, and its overall impact on assessment data should be discussed. ED G3.3 can be satisfied by providing an experimental error analysis.

3.4.4 ED G4—Test Conditions

The test conditions should be representative of prototypical conditions. Test specimens used in experiments should be representative of the proposed fuel design (i.e., the fuel design submitted for safety review). This goal is decomposed, as shown in Figure 3-17, into two supporting goals, which are discussed below.

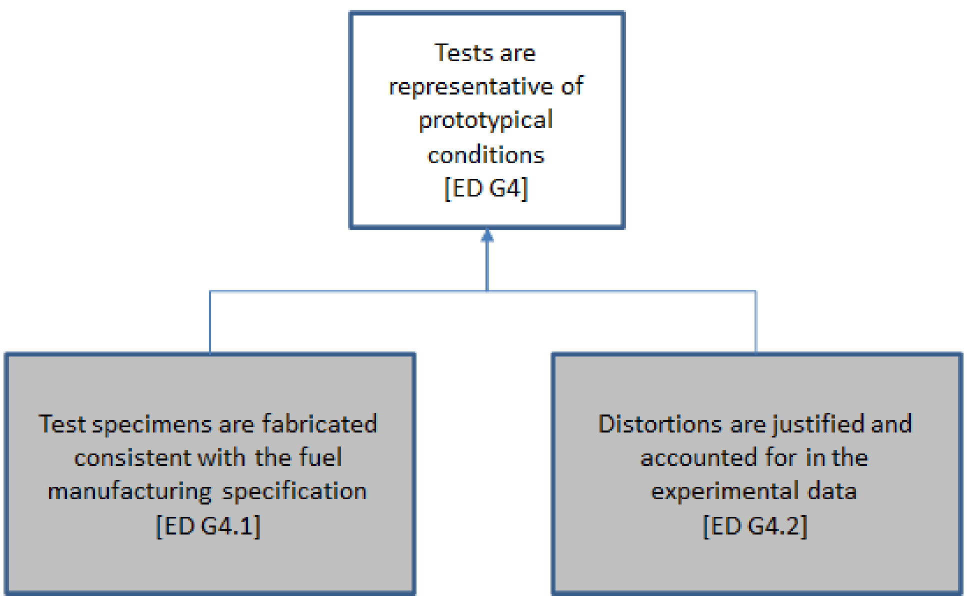

Figure 3-17 Decomposition of ED G4, "Test Specimens"

3.4.4.1 ED G4.1—Manufacturing of Test Specimens

Test specimens should be fabricated consistently with the manufacturing specification. (This goal is associated closely with G1, "Fuel Manufacturing Specification" (Section 3.1 of this report), which emphasized that fuel performance during normal operation and accident conditions can be highly sensitive to the fuel fabrication process.) It may be possible to provide justification for any acceptable differences in fabrication between the fuel and test specimens. Such justifications will be addressed case by case. ED G4.1 can be satisfied by demonstrating that test specimens are fabricated consistently with the fuel manufacturing specification.

3.4.4.2 ED G4.2—Evaluation of Test Distortions

Test distortions should be evaluated. Test distortions arise from differences between the test and the actual conditions under which the fuel is expected to perform (e.g., differences in test dimensions, initial and boundary conditions). An example of an expected test distortion is the geometry distortion typical of transient testing in a test reactor, as test reactors are generally too small to accommodate full-size fuel designs. ED G4.2 can be satisfied by an analysis of test distortions and justification for any identified distortions.

4 SUMMARY AND CONCLUSIONS

Section 3 of this report presents a systematic evaluation and justification of the ~~requirements~~acceptance criteria for qualifying nuclear fuel, and the table in Appendix A includes a concise list of the criteria identified to support a determination that nuclear fuel is qualified for use. These criteria provide a basis to support regulatory findings in the area of fuel qualification, as follows:

- The regulation in 10 CFR 50.43(e)(1)(i), requiring that the performance of each safety feature of the design has been demonstrated, is satisfied for the fuel by demonstrating that the safety criteria (G2 of the FQAF, discussed in Section 3.2 of this report) can be satisfied, which requires information to provide assurance that the fuel will perform as described in the safety analysis.

- The regulation in 10 CFR 50.43(e)(1)(iii) requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. This requirement can be satisfied by (1) specifying the fuel performance envelope, which covers a sufficient range of conditions (G2.1.1 of the FQAF), and (2) by demonstrating that assessed evaluation models and empirical criteria are capable of evaluating the fuel performance over the performance envelope (G2.1.2, G2.2.2, G2.2.3, G2.3.1, and G2.3.2(b) of the FQAF). Sections 3.2.1.1, 3.2.2.1, 3.2.2.3, 3.2.2.4, 3.2.3.1, and 3.2.3.2.2 of this report discuss these topics further.

- PDC similar to GDC 2 and ARDC 2 should specify~~require~~ that SSCs important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. G2.3 of the FQAF (discussed in Section 3.2.3 of this report) partially addresses this requirement through assurance of the ability to achieve and maintain safe shutdown.

- PDC similar to GDC 10 and ARDC 10 should specify~~require~~ that SAFDLS or SARRDLs not be exceeded under any conditions of normal operation, including the effects of AOOs. This requirement is satisfied, in part, by demonstrating margin to design limits under conditions of normal operation, including the effects of AOOs (G2.1 of the FQAF, discussed in Section 3.2.1 of this report).

- PDC similar to GDC 27 and ARDC 26 should specify~~require~~, in part, the ability to achieve and maintain safe shutdown under postulated accident conditions. G2.3 of the FQAF (discussed in Section 3.2.3 of this report) partially addresses this requirement through assurance of the ability to achieve and maintain safe shutdown.

- PDC similar to GDC 35 and ARDC 35 should specify~~require~~ an emergency core cooling system that provides sufficient cooling under postulated accident conditions. They also require that fuel and clad damage that could interfere with continued effective core cooling is prevented. G2.3 of the FQAF (discussed in Section 3.2.3 of this report) partially addresses these requirements through assurance of the ability to achieve and maintain safe shutdown.

- The regulations in 10 CFR 50.34(a)(1)((ii)(D), 10 CFR 52.47(a)(2)(iv), and 10 CFR 52.79(a)(1)(vi) require an evaluation of a postulated fission product release. This requirement is partially addressed by demonstrating margin to radionuclide release limits under accident conditions (G2.2 of the FQAF, discussed in Section 3.2.2 of this report).

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APPENDIX A
LIST OF ALL GOALS

As discussed in Section 3 of this report, the fuel assessment framework is developed using a top-down approach that starts with the high-level goal (G) that the fuel is qualified and then decomposes this goal into subgoals. Meeting the subgoals indicates that the higher-level goal is met. Each subgoal can either be further decomposed into other subgoals, or if no further decomposition is deemed necessary, the subgoal may be considered a base goal and evidence must be provided to demonstrate that the base goal is satisfied. In the following tables, base goals are identified by the use of grey boxes.

Table A-1 List of Goals in Fuel Qualification Assessment Framework

GOAL				**Fuel is qualified for use**
G1				Fuel is manufactured in accordance with a specification
	G1.1			Key dimensions and tolerances of fuel components are specified
	G1.2			Key constituents are specified with allowance for impurities
	G1.3			End state attributes for materials within fuel components are specified or otherwise justified
G2				Margin to safety limits can be demonstrated
	G2.1			Margin to design limits can be demonstrated under conditions of normal operation and AOOs
		G2.1.1		Fuel performance envelope is defined
		G2.1.2		Evaluation model is available (see EM Assessment Framework)
	G2.2			Margin to radionuclide release limits under accident conditions can be demonstrated
		G2.1.1		Fuel performance envelope is defined
		G2.2.1		Radionuclide retention requirements are specified
		G2.2.2		Criteria for barrier degradation and failure are suitably conservative
			(a)	Criteria are conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
		G2.2.3		Radionuclide retention and release from fuel matrix are modeled conservatively
			(a)	Model is conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
	G2.3			Ability to achieve and maintain safe shutdown is assured
		G2.3.1		Coolable geometry is ensured
			(a)	Criteria to ensure coolable geometry are specified
			(b)	Evaluation models are available (see EM Assessment Framework)
		G2.3.2		Negative reactivity~~Control element~~ insertion can be demonstrated
			(a)	Criteria are provided to ensure that negative reactivity~~control element~~ insertion path is not obstructed
			(b)	Evaluation model is available (see EM Assessment Framework)

Table A-2 List of Goals in Evaluation Model Assessment Framework

GOAL	Evaluation model is acceptable for use		
EM G1	Evaluation model contains the appropriate modeling capabilities		
	EM G1.1	Evaluation model is capable of modeling the geometry of the fuel system	
	EM G1.2	Evaluation model is capable of modeling the material properties of the fuel system	
	EM G1.3	Evaluation model is capable of modeling the physics relevant to fuel performance	
EM G2	Evaluation model has been adequately assessed against experimental data		
	EM G2.1	Data used for assessment are appropriate (see ED Assessment Framework)	
	EM G2.2	Evaluation model is demonstrably able to predict fuel failure and degradation mechanisms over the test envelope	
		EM G2.2.1	Evaluation model error is quantified through assessment against experimental data
		EM G2.2.2	Evaluation model error is determined throughout the fuel performance envelope
		EM G2.2.3	Sparse data regions are justified
		EM G2.2.4	Evaluation model is restricted to use within its test envelope

Table A-3 List of Goals in Experimental Data Assessment Framework

GOAL	Experimental data used for assessment are appropriate	
ED G1	Assessment data are independent of data used to develop/train the evaluation model	
ED G2	Data has been collected over a test envelope that covers the fuel performance envelope	
ED G3	Experimental data have been accurately measured	
	ED G3.1	The test facility has an appropriate quality assurance program
	ED G3.2	Experimental data are collected using established measurement techniques
	ED G3.3	Experimental data account for sources of experimental uncertainty
ED G4	Test specimens are representative of the fuel design	
	ED G4.1	Test specimens are fabricated consistent with the fuel manufacturing specification
	ED G4.2	Distortions are justified and accounted for in the experimental data

APPENDIX B
ANALYSIS OF PUBLIC COMMENTS

Comments on the subject draft NUREG-2246 are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can access the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of the NRC's public documents. The following table lists the comments the NRC received on the draft NUREG.

Table B-1 Public Comment Submittals on Draft NUREG-2246

Letter Number	ADAMS Accession No.	Commenter Affiliation	Commenter Name
1	ML21243A353	Public	Anonymous
2	ML21243A356	Nuclear Energy Institute (NEI)	Ben Holtzman
3	ML21246A124	Public	Aleksey Rezvoi

The original comment as written by the commenter in its letter above is listed first, followed by the NRC staff's response.

Letter 1—Public

<u>Comment No. 1-1</u>

Do you know if any Republicans will be involved in testing nuclear fuel? The reason I ask is that many Republicans are anti-science, anti-government, fact challenged conspiracy theorists. And yes, this does affect their work.

A recent example is county clerk in Mesa country, CO. When her county's voting machines were due to be serviced she shut off surveillance camera and snuck in an unauthorized technician. He stole passwords and an image of the hard drive, which were given to right wing conspiracy websites and the My Pillow guy's cyber conference. Now she's being investigated by the FBI, and Mesa county has to buy new voting machines since these ones have been compromised.

So entrusting a rightwing nutjob test nuclear fuels would be like counting on President Tiny Hands to run a pandemic response. In other words, a disaster.

<u>NRC Response</u>

The fuel qualification framework provides criteria derived from regulatory requirements that, when satisfied, would support regulatory findings that a nuclear fuel is qualified. As long as NRC regulations are met, regardless of the political affiliations of those involved in testing nuclear fuel, adequate protection of the public health and safety is assured.

The NRC staff made no changes to NUREG-2246 in response to this comment.

Letter 2—NEI

<u>Comment No. 2-1</u>

General

There are numerous advanced reactor designs, with many different fuel designs, and each fuel design has its own unique challenges and advantages. This draft NUREG has a lot of guidance heavily based on existing LWR fuel designs, which may not be applicable to all advanced reactor designs. Please make the guidance more applicable to non-traditional fuel designs.

Please add examples for non-traditional LWR fuel designs.

NRC Response

NRC staff acknowledges that NUREG-2246 is informed by lessons-learned from experience with light water reactor fuel (in the US and internationally) but disagrees with the characterization that it is heavily based on light water reactor fuel designs. Section 1.4 of NUREG-2246 clarifies that:

> *The assessment criteria in Section 3 (referred to as goals) of this report draw on regulatory experience gained from licensing solid fuel reactor designs (particularly [light water reactor] LWR designs), results from advanced reactor fuel testing performed to-date, and accelerated fuel qualification (AFQ) considerations. An attempt has been made to develop a generically applicable set of base goals. However, some goals may not apply to all fuel types or reactor designs (e.g. molten salt reactor fuel) …*

Advanced reactor designs have been considered throughout the document and only a small subset of the reference material for NUREG-2246 is specific to traditional LWR fuel.

NRC staff does not have sufficient information to provide example criteria for specific advanced reactor fuel types. However, the NRC staff has initiated separate activities with Oak Ridge National Laboratory on molten salt reactor fuel qualification, and NRC has awarded contracts to Idaho National Laboratory and Pacific Northwest National Laboratory to exercise the framework outlined in NUREG-2246 through generic assessments of metal fuel and tristructural isotropic (TRISO) fuel. These assessments are not needed to describe the generic framework presented in the NUREG but could be incorporated into the NUREG in a future revision.

Comment No. 2-2

General

Industry believes the draft NUREG allows for the flexibility for accelerated fuel qualification compared to the historical 20-year duration.

None

NRC Response

AFQ considerations are incorporated into NUREG-2246 as described in Section 2.3 of the report. The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-3

General

The term "safety case" appears to describe the safety function of the fuel independent of the definition from using risk-informed performance-based methodologies.

Please change "Safety Case" to "Fuel Safety Functions" in the document.

<u>NRC Response</u>

The NRC staff agrees that the term “safety case” is undefined. This term is eliminated from Section 2.2.4, Section 3.2, and Section 3.2.2.2 of the report. Instead, the NRC staff presents this concept as the role of nuclear fuel in the protection against the release of radioactivity. “Safety case” is used in NUREG-2246 only (1) when quoting material that uses it, and (2) as a title in Section 1.3 of the report (previously Section 1.2). As described in footnote 2 (previously footnote 3), “Safety Case” is used as the title to Section 1.2 of the report because it is a term frequently encountered in literature that addresses fuel qualification.

<u>Comment No. 2-4</u>

Page iii Line 4

The text states that "The purpose of this report is to identify criteria that will be useful for advanced reactor designs through an assessment framework that would support regulatory findings associated with nuclear fuel qualification." It is important to note that there is no specific regulatory requirement for the "qualification of fuel." The regulatory requirement referenced in this NUREG is 50.43, which requires the "performance of novel safety features be demonstrated through either analysis, appropriate test programs, experience, or a combination thereof." Fuel qualification is therefore only necessary to meet the requirements in Section 2.1 to the extent that the fuel is specifically relied upon as a safety feature.

The abstract section should be revised to clarify that fuel qualification is not a regulatory requirement and provide more context to how potentially applicable requirements would inform what is necessary for fuel qualification.

<u>NRC Response</u>

The NRC staff agrees that the regulatory need for fuel qualification needs to be described but disagrees that this level of detail should be included in the abstract. Section 2.1, “Regulatory Basis” of NUREG-2246 includes the requested information (i.e., applicable requirements) and clarifies that:

> *The fuel qualification assessment framework in Section 3 of this report provides guidance to facilitate an efficient and transparent licensing review in the area of fuel qualification. The guidance provided in this report is not a substitute for the Commission’s regulations, and compliance with the guidance is not required.*

The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-5

Page 1-1 Line 6

The NUREG primarily references the Crawford report when establishing the objective of fuel qualification and its corresponding basis. A technical paper (the Crawford report) does not establish a regulatory basis and is not appropriate to establish the regulatory need for fuel qualification. Furthermore, the noted economic operation in the Crawford report is not something that NRC should be regulating on.

Please replace the reference of a technical paper with a regulatory one (when identified as a means to do so) and remove reference to economics as a regulatory goal.

NRC Response

The NRC staff disagrees with the characterization that a technical paper served as the primary basis for NUREG-2246, or that the NRC is regulating on the economic considerations discussed in the referenced technical paper. The regulatory basis for fuel qualification is described in Section 2.1 of NUREG-2246 which states, "Nuclear fuel qualification to support reactor licensing involves the development of a basis to support findings associated with regulatory requirements that apply to the nuclear facility. This section discusses these requirements and their relationship to this report."

The NRC staff uses relevant literature to inform their judgement when applicable and acknowledges that the research by Crawford, Porter, Hayes, Meyer, Petti, and Pasamehmetoglu informed their judgement on an applicable definition of "fuel qualification," for use in NUREG-2246. However, the definition of fuel qualification used in NUREG-2246 was selected by the NRC staff because the NRC staff determined that is an accurate characterization of fuel qualification based on the NRC regulations, NRC staff's experience from licensing solid fuel reactor designs, results from advanced reactor fuel testing performed to-date, and AFQ considerations.

The NRC staff agrees that the research of Crawford, Porter, Hayes, Meyer, Petti, and Pasamehmetoglu can be better characterized in NUREG-2246 to prevent misunderstanding regarding the use of the research literature in the development of regulatory guidance. The following revisions will be made to NUREG-2246:

<u>The first paragraph of Section 1.1, "Purpose," will be revised as follows:</u>

The purpose of this report is to provide a fuel qualification assessment framework for use with advanced reactor designs, including power and non-power reactors, that addresses regulatory requirements. Specifically, the framework provides criteria (referred to as base goals) derived from regulatory requirements that, when satisfied, would support regulatory findings that a nuclear fuel is qualified. This report provides the bases for the identified base goals and clarifying examples for the types of information that an applicant should provide for the NRC to determine that these goals are satisfied, and regulatory requirements are met. Appendix A lists all goals within the framework.

<u>Section 1.2, "Definitions" is added as follows:</u>

The term “fuel qualification” is not explicitly defined or used in NRC regulations. However, there are regulatory requirements applicable to power reactor applications that are generally associated with nuclear fuel behavior under conditions of normal operation, anticipated operational occurrences (AOOs), and accident conditions (see Section 2.1 of this report). The information needed to understand fuel behavior under these conditions is generally obtained as part of a fuel qualification process. Research literature provides additional insight into the objectives of fuel qualification from the developers’ points of view (Crawford, et. al., 2007), (Terrani, et. al., 2020) which highlight the needs to (1) fabricate a fuel product in accordance with a specification, (2) meet licensing safety-requirements, and (3) meet reliability needs. This report uses the following definitions of “qualified fuel” and “fuel qualification” as an aid to NRC staff in the development of the fuel qualification framework.

> “Qualified fuel” means fuel for which reasonable assurance exists that the fuel, fabrication in accordance with its specification, will perform as described in the safety analysis.
>
> “Fuel qualification” means the overall process (planning, testing, analysis, etc.) used to obtain qualified fuel.

These definitions were used because they characterize information required by NRC regulations as described in Section 2.1, “Regulatory Basis” and are consistent with the NRC staff’s experience from licensing solid fuel designs (particularly light water reactor (LWR) fuel), advanced reactor fuel testing performed to-date, and AFQ considerations.

Sections 1.2 and 1.3 are renumbered to 1.3 and 1.4.

Section 3 is updated to incorporate the definition from Section 1.2.

Comment No. 2-6

Page 1-1 Line 14

The term "base goal" is confusing - even with the footnote.

Please add additional information regarding how the base goal meets high-level regulatory requirements.

NRC Response

The NRC staff appended the footnote to clarify that this is further described in Section 3 of the report.

Footnote 1 updated revised to, “A base goal is a goal that is not decomposed any further but is supported by evidence. The top-down approach and decomposition of regulatory requirements into base goals is described in Section 3.”

Comment No. 2-7

Page 1-1 Lines 20-24

The text states "this framework relies on regulatory requirements that are applicable to applications for design certifications, combined licenses, manufacturing licenses, or standard design approvals." However, fuel qualification itself is not required for any of the mentioned licensing approvals. Fuel qualification is therefore only necessary to meet the requirements in Section 2.1 to the extent that the fuel is specifically relied upon as a safety feature.

Please clarify the connection of fuel qualification to the noted licensing approvals, or modify the text to more clearly denote more context to how potentially applicable requirements would inform what is necessary for fuel qualification.

NRC Response

The NRC staff recognizes that the term “fuel qualification” is not explicitly defined or used in NRC regulations. However, there are regulatory requirements generically applicable to power reactor applications that are generally associated with nuclear fuel behavior and its role in preventing the release of radioactive material under conditions of normal operation, including the effect of anticipated operational occurrences and accident conditions. These regulatory requirements and their connection to the fuel qualification framework are identified and described in Section 2.1 of NUREG-2246, “Regulatory Basis.”

The NRC staff further recognizes that safety functions attributed to nuclear fuel, and the means of accomplishing these safety functions, can vary depending on the reactor design. However, the fundamental safety functions of reactivity control, heat removal, and radionuclide retention are generally associated with nuclear fuel. The regulatory basis provided in Section 2.1 and the fuel qualification assessment framework was specified, in part, to address these fundamental safety functions.

Additionally, Section 2.1 of NUREG-2246 also states, “The fuel qualification assessment framework in Section 3 of this report provides guidance to facilitate an efficient and transparent licensing review in the area of fuel qualification. The guidance provided in this report is not a substitute for the Commission’s regulations, and compliance with the guidance is not required.”

Comment No. 2-8

Page 1-2 Lines 4-6

The following text is confusing because it is not consistent with the stated purpose of the document: "The scope of this report focuses on the identification and understanding of fuel life-limiting failure and degradation mechanisms due to irradiation during reactor operation." Therefore, it is unclear whether the purpose of the report is for the identification of safety functions of the fuel and its performance in a reactor, or for fuel performance in general.

Please clarify the regulatory basis of this NUREG and ensure alignment throughout the report.

NRC Response

The NRC staff disagrees that the scope statement is inconsistent with the stated purpose. Section 1.1, “Purpose,” of NUREG-2246 states, “The purpose of this report is to provide a fuel qualification assessment framework for use with advanced reactor designs that satisfies regulatory requirements,” and Section 1.4, “Scope” of NUREG-2246 (previously Section 1.3) clarifies that fuel affects other aspects of nuclear safety that are beyond the scope of this report.

Additionally, Section 2.1, “Regulatory Basis,” provides the regulatory basis associated with fuel qualification. Both Section 2.1, “Regulatory Basis” and Section 4, “Summary and Conclusions” of the report provide the connection between the regulations and the goals within the fuel qualification framework which provides confirmation of alignment throughout the report. Draft NUREG-2246 was reviewed by the Office of General Counsel (OGC) prior to publication in the Federal Register.

The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-9

Page 1-2 Lines 5-6 and Section 3.1.1

Fuel life-limiting failure and degradation mechanisms are not just due to irradiation during reactor operation. Other degradation mechanisms, chemical attacks, hydrogen pickup, high temperature, and time at temperature during AOOs or Design Basis Accidents also impact fuel performance.

Please remove "due to irradiation" to expand the applicability of the statement to include other failure mechanisms.

NRC Response

In response to this comment, the NRC staff changed “due to irradiation” to “due to irradiation and irradiation assisted phenomena.” The NRC staff agrees that there are degradation mechanisms beyond irradiation. However, the primary obstacle to qualifying nuclear fuel has generally been demonstrating fuel performance at the desired exposure and most fuel degradation phenomena are impacted by irradiation.

Comment No. 2-10

Section 2.1

The regulatory basis denoted is 50.43(e) and the design criteria (GDC and ARDC). However, the GDC and ARDC are not requirements for, and as guidance are not required to be met by, non-LWRs. Thus, non-LWRs may choose to develop PDCs through another method. The guidance should clarify that fuel qualification is only necessary if it is determined to be one of the PDCs for the design, based upon the fuel being relied upon as a safety feature.

Please clarify the text to indicate how fuel qualification could be used to demonstrate compliance to 50.43 but is not necessary if fuel is not relied upon as a safety feature. Additional context on how potentially applicable requirements would inform what is necessary for fuel qualification would be helpful.

NRC Response

The NRC staff agrees that GDCs and ARDCs are not requirements for non-LWRs but are instead considered guidance for non-LWR advanced reactor applicants in developing proposed principal design criteria (PDCs). PDCs are required to be proposed by applicants in accordance with the following regulations: 10 CFR 50.34; 10 CFR 52.47, 52.79, 52.137, and 52.157. An

applicant should propose PDC associated with the fuel design for their specific design when they are necessary to establish design and performance requirements that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Such PDC may be deemed necessary by the NRC staff. Applicants for LWRs, including small modular LWRs, may be required to meet the GDC in Appendix A to 10 CFR Part 50 for fuel design and qualification. Applicants for small modular LWRs should take advantage of pre-application interactions with the NRC staff to ensure that their approach to fuel qualification and associated proposed PDCs is consistent with the intent of the regulations.

In response to this comment, the NRC staff is updating Section 2.1 of NUREG-2246 to accurately reflect requirements associated with principal design criteria.

Comment No. 2-11

Section 2.2.1

The specific purpose of the guidance is not clear throughout the document. The guidance discusses both fuel qualification and how to prevent mechanical damage of traditional LWR fuel - as noted in Section 2.2.1.

Please provide additional clarification regarding the intent of the guidance.

NRC Response

In response to this comment and other comments related to the purpose of NUREG-2246, the NRC staff added clarifying text in Section 1.1 of the report. The purpose of NUREG-2246 is to provide a fuel qualification assessment framework for use with advanced reactor designs that addresses regulatory requirements. Draft NUREG-2246 and the preceding documents upon which it is based have already undergone a significant amount of review and incorporation of stakeholder input.

Section 2.2.1, "NUREG-0800, Section 4.2" of NUREG-2246 does discuss relevant guidance applicable to traditional LWR fuel (including the prevention of mechanical damage), but also highlights that the guidance in NUREG-0800 may not apply or may not suffice to address advanced reactor technologies that use different fuel forms, or address situations in which the fuel plays different roles in the protection against the release of radionuclides.

Comment No. 2-12

Page 2-3 Line 17

Section 3.2.2 refers to "fuel failures" rather than "fuel rod failures" as noted in this paragraph. Please consider using "fuel failures" as it is more generally applicable. NUREG-0800 appropriately uses the term "fuel rod failures."

Please replace "fuel rod failures" with "fuel failures."

NRC Response

In response to this comment, the NRC staff changed "fuel rod failures" to "fuel failures."

Comment No. 2-13

Section 2.2.2

While the information listed here relates to Accident Tolerant Fuel (ATF), which may only be applicable to some advanced reactor fuel designs, the intent of the section appears to be the potential of the PIRT process. It is therefore confusing to title the section as the ATF ISG.

Please revise this section to focus on the PIRT process as one acceptable method for identifying failure mechanisms and features of an evaluation model.

NRC Response

The NRC staff disagrees with the comment that the tile is confusing because the title of Section 2.2 is "Related Guidance" and "ATF-ISG-2020-01" is a summary title for the guidance document "ATF-ISG-2020-01 Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept."

However, in response to this comment and to provide further clarification, the NRC staff changed the last sentence in Section 2.2.2 to:

However, as discussed in the evaluation model assessment framework in Section 3.3 of this report, the PIRT process may be used to identify failure mechanisms and necessary features of an evaluation model.

Comment No. 2-14

Page 2-4 Lines 18-20

The following text is misleading because fuel may have safety functions as noted in the text, but it is not required to. It is possible to not credit the fuel and instead credit other mechanisms outside of the fuel matrix. Fuel qualification is therefore only necessary to meet the requirements in Section 2.1 to the extent that the fuel is specifically relied upon as a safety feature.

"Fuel qualification partially addresses the fundamental safety functions of control of reactivity, cooling of radioactive material, and confinement of radioactive material…"

Please clarify the role of fuel qualification and its necessity only if being relied upon and/or credited in the safety analysis as some designs may not. Additional context on how potentially applicable requirements would inform what is necessary for fuel qualification would be helpful.

NRC Response

The NRC staff recognizes that the role of fuel in the protection against the release of radioactivity can vary depending upon the reactor design (see Section 1.2, "Safety Case" in NUREG-2246 (this is Section 1.3 in the updated NUREG)). The fuel qualification framework, and it's associated regulatory basis, was developed to be as generic as possible. For example, Section 3.2.2.2 of this report, "G2.2.1 – Radionuclide Retention Requirements" recognizes that the degree of radionuclide retention credited to occur within the fuel can vary widely between different reactor designs. The associated goals under G.2.2, "Margin to radionuclide release

limits under accident conditions can be demonstrated" can be addressed in a graded manner in accordance with the degree to which the fuel is credited.

To clarify the relationship between NUREG-2246 and RG 1.233, the second paragraph of Section 2.3.3 is revised to the following:

Additionally, Regulatory Guide 1.233 discusses fundamental safety functions. Nuclear fuel contributes to the reactivity balance and is a source of heat generation and fission products. Therefore, nuclear fuel is generally recognized as impacting the fundamental safety functions of reactivity control, heat removal, and confinement of radioactive material. Fuel qualification may partially addresses these fundamental safety functions by incorporating the role of the fuel in these safety functions in G2, "Safety Criteria," which is discussed in Section 3.2 of this report, as follows:

- Confinement of radioactive material is partially addressed by G2.1, "Design Limits during Normal Operation and AOOs," and G2.2, "Radionuclide Release Limits." These goals allow for a graded approach such that information should be provided in accordance with the degree to which the fuel is credited in the protection against the release of radionuclides.

- Control of reactivity and cooling or radioactive material are partially addressed by G2.3, "Safe Shutdown." This goal allows for a graded approach such that information should be provided in accordance with the degree to which the fuel can obstruct the insertion of negative reactivity.

Comment No. 2-15

Page 2-4 Lines 31-35

The text mentions work on fuel qualification for molten salt reactors, but does not mention the work on other fuel designs such as EPRI's Topical Report for TRISO fuel qualification.

Please add information for the TRISO fuel qualification and other relevant development work.

NRC Response

The NRC staff disagrees with the comment regarding the addition of a reference to the TRISO fuel qualification work to Section 2.2.4 of the report, "Guidance in Development," in NUREG-2246 because discussion of TRISO fuel is not applicable to that specific section. NUREG-2246 references EPRI-AR-1(NP)-A, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance," and other reports related to advanced reactor fuel in other sections within the report. The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-16

Section 2.4 Page 2-5 and Section 3.4.2 Page 3-17, Lines 9-20

The text here on lead test specimen programs is applicable only if we had existing/operating advanced reactors. It does not discuss alternatives for fuel to be qualified for first core applications where lead test specimens are not possible.

Please add information on fuel qualification for first core applications.

NRC Response

In response to this comment, the NRC staff added Section 2.5, "First Core Applications" (and renumbered the current Section 2.5 to 2.6 in the report).

2.5 First Core Applications

Nuclear fuel contributes to the reactivity balance and is a source of heat generation and fission products. Therefore, it is generally recognized as impacting the safety functions of reactivity control, heat removal, and confinement of radioactive material. Accordingly, nuclear fuel would be considered a safety feature subject to the requirements of 10 CFR 50.43(e) and would require demonstration prior to licensing.

Section 2.1 identifies the requirements under 10 CFR 50.43(e)(1) to demonstrate the performance of each safety feature of a design and to have sufficient data on the safety feature to assess analytical tools. The assessment framework in Section 3 of this report is developed to address the requirements identified in Section 2.1, including 10 CFR 50.43(e)(1)(i) and 10 CFR 50.43(e)(1)(iii). As described in Section 2.4 and Section 3.4.2, the framework provided in Section 3 accommodates the use of a lead test specimen program beyond what has been traditionally used for LWRs. However, to address the regulatory requirements identified in Section 2.1, sufficient data on the safety features of the fuel is necessary. This data may be obtained from test reactors provided that the tests and associated data are appropriate for assessing evaluation models (see Section 3.4).

As an alternative to the requirements of 10 CFR 50.43(e)(1), 10 CFR 50.43(e)(2) allows the use of a prototype plant to comply with testing requirements. To date, the NRC has not licensed a nuclear power plant as a prototype, however, it has been envisioned that this could be a pathway for licensing advanced reactors (see Appendix B to Enclosure 1 of the Regulatory Roadmap (NRC, 2017)). Use of a prototype plant may involve the application of additional NRC imposed requirements on siting, safety features, or operational conditions to protect the public and the plant staff from the possible consequences of accidents during the testing period.

NRC imposed requirements under 10 CFR 50.43(e)(2) may appear as license conditions. Any imposed conditions would consider risk-insights and any uncertainties associated with fuel performance that are to be addressed by testing in the prototype reactor. Furthermore, the adequacy of these license conditions would be subject to the mandatory hearing (performed as part of plant licensing) and NRC staff expects license conditions to be an area of significant focus during a detailed technical review. Due to the significant impact that such a licensing strategy would have on the overall safety review of the facility, NRC staff encourages any applicant considering such a licensing strategy to have significant pre-application engagement on the topic of fuel qualification (NRC, 2021) to ensure that there is a common understanding of associated technical, regulatory, and policy issues and to reduce regulatory uncertainty.

References

NRC. (2017). "A Regulatory Review Roadmap for Non-Light Water Reactors," December 2017(ADAMS Accession No. ML17312B567).

NRC. (2021). "Draft Pre-application Engagement to Optimize Advanced Reactors Application Reviews," May 2021 (ADAMS Accession No. ML21145A106).

Comment No. 2-17

Section 3.1.3

The TRISO SER allows for manufacturing independence as long as the final product has properties that fall within the specification range. Please note TRISO as an example of "insensitivity to manufacturing processes."

Please add text to denote that the TRISO is an example of fuel that has an insensitivity to manufacturing processes and instead measurable criteria can be used to justify predicted performance.

NRC Response

In response to this comment, the NRC staff revised Section 3.1 and 3.1.3 to the following:

Section 3.1:

Fuel performance during normal operation and accident conditions can be highly sensitive to fuel manufacturing parameters. For example, failure criteria during reactivity-initiated accidents for LWRs with zirconium-based cladding depend upon the heat treatment of the cladding because of its impact on microstructure (NRC, 2020c). Similarly, key manufacturing parameters for TRISO particles have been identified that provide assurance of fuel performance during normal operation (EPRI, 2020).

Section 3.1.3

End state attributes (e.g. microstructure) for the materials within fuel components should be specified or otherwise justified. The information necessary to capture the desired end state of the material may take several forms. For example, the identification and justification of key end-state parameters has been an acceptable means for providing assurance for TRISO particles for performance under conditions of normal operation (EPRI, 2020). Alternatively, specific manufacturing processes (e.g., cold-working, heat treatments, acid pickling, deposition techniques) that are essential to create the microstructure may be indicated in lieu of end state attributes. In some cases, it may be preferable to use performance-based end state attributes that can be supported through periodic testing and reporting (NRC, 2016). Additionally, it may be possible to demonstrate insensitivity to manufacturing processes so that end state attributes need not be specified in licensing documentation. Licensing documentation should provide sufficient justification for cases where a specific material is insensitive to manufacturing processes.

Comment No. 2-18

Figures 3-3 and 3-8, Section 3.2.3 and other locations

Criterion G2.3 includes a statement for the "ability to achieve and maintain safe shutdown can be assured." The NUREG further defines safe shutdown as "a state in which the reactor is subcritical, decay heat is being removed, and radionuclide inventory is contained."

However, safe shutdown and safe state are not interchangeable. Not all reactors must be subcritical to be safe and therefore it is not necessary for all fuel types to be subcritical. Industry recommends aligning the NUREG with other documentation that use the phrase "a safe stable end-state" instead.

Please change criterion G2.3 to "Ability to achieve and maintain a safe, stable, end state" and revise the text throughout to be consistent with this revised criterion.

NRC Response

The NRC staff agrees that the phrase “a safe stable end-state” is used in other documentation. Relatively recent NRC policy papers have clarified that maintaining subcriticality with only safety-related structures, systems, and components may not be required (SECY-18-0099). However, NRC staff continues to expect that nuclear fuel be designed such that forces on the nuclear fuel, resulting from internal or external events, will not preclude the eventual achievement of a subcritical state. The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-19

Page 3-4 Line 7

The text states that "This performance envelope informs the safety analysis and technical specifications for the design (i.e., limiting conditions for operation)." The use of LCO in this context is unclear. LCOs are typically defined for active systems in a facility, whereas design features are defined for fuel systems.

Please remove the reference to LCOs or clarify how this is intended to inform LCO development.

NRC Response

In response to this comment, the NRC staff updated the text to state, “This performance envelop informs the safety analysis, technical specifications and/or operating limits for the design.” In addition, the NRC staff added a footnote to clarify the connection to 10 CFR 50.36(c)(2)(ii) and associated requirements in 10 CFR Part 52.

10 CFR 50.36(c)(2)(ii)(B) Criterion 2 requires a limiting condition for operation for a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Comment No. 2-20

Page 3-5 Line 20

The statement that "conditions to which the fuel is subjected during design-basis accidents may be specified independent of the reactor design…" is unclear and should be further justified.

Please clarify the text.

NRC Response

Note the preceding statement, “as discussed in Section 3.2.1.1.” Section 3.2.1.1, “G2.1.1-Definition of Fuel Performance Envelope,” provides additional background on this goal. In response to the comment, the NRC staff added the following clarifying language to Section 3.2.2.1:

For example, bounding conditions, (e.g., temperatures, pressures, transient power) may be provided that would encompass the performance of an unspecified reactor design.

Comment No. 2-21

Page 3-6 Lines 1-2

The following text is vague. "To satisfy G2.2.1, the degree of radionuclide retention within the fuel system should be specified."

Please clarify the text regarding the quantification of retention.

NRC Response

In response to this comment, the NRC staff changed the text as follows: “To satisfy G2.2.1, the radionuclide retention requirements of the fuel system under accident conditions should be specified.”

Comment No. 2-22

Page 3-6 Lines 19-20

The text denoting the statistical confidence level example "(e.g., 95/95)" is one way to incorporate uncertainties. Another way is to bound it within the uncertainty quantification parameters.

None

NRC Response

In response to this comment, the NRC staff changed the text as follows: “Ideally, sufficient data would be available to demonstrate margin with uncertainty at a statistical confidence level using an independent set of data.”

Comment No. 2-23

Page 3-8 Lines 14-16

No specific criterion is set as to what the term "coolable geometry" means for non-traditional LWR fuel designs. For example, some advanced reactor fuel designs like the MSFR and the MCFR are planning on using liquid salt with fissile product as both their coolant and their fuel, which would require clarification/flexibility on the definition of "coolable geometry."

Please clarify what a coolable geometry criterion would be for fuel designs that do not have a containment (e.g., LWR fuel cladding) and how this will ensure ability to attain safe, stable, end state.

NRC Response

Section 3.2.3.1.1, "G2.3.1(a) – Identification of Phenomena" of NUREG-2246 provides examples of the types of phenomena that could cause a loss of coolable geometry. These phenomena and associated coolable geometry criteria cannot be specified generically for all fuel designs and would need to be provided by an applicant to address their specific fuel design. Section 1.4, "Scope," of NUREG-2246 (previously Section 1.3) clarifies that, "some goals may not apply to all fuel types or reactor designs (e.g., molten salt reactor fuel), and additional or alternate criteria should be applied (see Section 2.2.4 for guidance on molten salt reactor fuel)."

Comment No. 2-24

Page 3-9 Lines 2-8

Please include additional examples applicable for non-traditional LWR fuel designs (e.g., gas cooled reactors or solid core block reactor fuel designs). These designs may have fuel channels that are separate or away from control rods or drums. Noting diverse phenomena for such designs will make the NUREG more useful to industry applicants.

Please add examples for non-traditional LWR fuel designs.

NRC Response

NRC does not have sufficient information to provide specific examples for non-traditional LWR fuel designs. NUREG-2246 is intended to provide generic guidance and the specific criteria for non-LWR fuel needs to be provided and justified by the applicant. See the response to Comment 2-1 for additional details regarding on-going work related to non-traditional LWR fuel designs. The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-25

Page 3-9 Line 24

"form" should be "from"

Please made editorial change as noted.

NRC Response

The NRC staff agrees with the comment and changed "form" to "from" in response to this comment.

Comment No. 2-26

Page 3-10 Lines 5-9

For designs that do not have neutron control element insertion (e.g., control drums), having criteria to ensure control element insertion paths would not be necessary.

Please revise text to indicate criteria should be specified only for designs with neutron control elements whose insertion is credited in accident response models.

NRC Response

In response to this comment, Section 3.2.3, “G2.3-Safe Shutdown,” of NUREG-2246 was updated to replace “control element insertion” with “negative reactivity insertion.”

Additionally, the following text was added to Section 3 of NUREG-2246:

As discussed in Section 1.4, the assessment framework draws on regulatory experience gained from licensing solid fuel reactor designs (particularly LWR designs), results from advanced reactor fuel testing performed to-date, and AFQ considerations. An attempt has been made to develop generically applicable set of base goals. However, it is recognized some goals may not be applicable to all fuel types or reactor designs.

The NRC staff notes that the use of control drums, as provided in the commenters example, does not necessarily ensure the ability to insert negative reactivity.

Comment No. 2-27

Page 3-12 Lines 28-33

Some phenomena do not have a known explicit mechanistic physics code available. Please clarify whether evaluation physics models can be mechanistic or empirical in nature, so that phenomena that are known but do not have an explicit first-principles model can be covered. For example, a SiC layer degradation mechanism would not be explicitly modeled in fuel performance codes.

Please clarify the text regarding the modeling of phenomena.

NRC Response

The NRC staff agrees that evaluation models can be mechanistic or empirical. Section 3.3 of NUREG-2246 already contains the requested clarification in stating, “The term ‘evaluation model’ here is used in the generic sense. Typically, an evaluation model is an analytical tool, a computer code, or a combination of such tools. However, the use of a sophisticated tool such as a computer code may not be necessary to evaluate fuel performance. For example, a simple mathematical expression or set of data can serve as an evaluation model, if sufficient evidence exists to support its use.” The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-28

Page 3-16 Line 18

"Data analytics" may be more applicable than "machine learning" as the latter implies the need for an artificial intelligence system, which may not be needed to perform the role noted.

Please replace "machine learning" with "data analytics."

NRC Response

In response to this comment, the NRC staff struck the term “machine learning.” The revised sentence is:

Methods exist for determining whether the selection of the training data affects the resulting uncertainty; such methods include random subsampling and k-fold cross-validation.

Comment No. 2-29

Page 3-16, Lines 38-47

The text does not address methods for justifying when fuel can be used beyond its performance envelope when lead test specimens are not available.

Please add information on what adequate justification is needed to expand the performance envelope using experimental data without the use of lead test specimens.

NRC Response

The NRC staff is unable to provide generic guidance to address all potential scenarios where fuel use beyond its performance envelope may be requested. However, some information related to the use of an evaluation model outside of its test envelope is provided in Section 3.3.2.2.4, “EM G2.2.4-Restricted Domain,” of NUREG-2246 which states:

> *Application of an evaluation model outside of the supporting test envelope (see Section 3.4.2) may be justified based on physical arguments (e.g., that the evaluation model provides a simplified or bounding treatment of physical phenomena). Justification for extrapolation of a model outside of the test envelope is strengthened by the use of physics-based models, such as those discussed in Section 2.3, which are informed by fundamental information about fuel evolution and behavior, as opposed to empirically derived models (Terrani, et al., 2020).*

Additionally, in response to comment 2-16, the NRC staff added Section 2.5, “First Core Applications,” to NUREG-2246, which addresses the scenario where data is not available to address the regulatory requirements of 10 CFR 50.43(e)(i) and 10 CFR 50.43(e)(iii).

The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 2-30

Page 3-17 Line 34

ASME NQA-1 is not the only way to qualify fuel data. For data collected at national laboratories, the application of this standard may not be possible, despite the national laboratories using alternative and acceptable quality assurance methods. NRC should accept data from the technical experts at national laboratories if the data was collected under the lab's QA program

as noted in ML20054A297, where NRC staff determined that Argonne National Lab's quality assurance program plan is based on the method provided in ASME’s NQA-1-2008/2009 and satisfies the quality assurance requirements of Appendix B to 10 CFR Part 50.

Please either remove or modify the text to allow for data to also be qualified under the commercial grade dedication (CGD) process rather than stating data must be made compliant.

NRC Response

In response to this comment, the NRC staff revised Section 3.4.3.1 to state the following:

Experimental data should be collected under an appropriate quality assurance program that meets applicable regulatory requirements. Standards such as the American Society of Mechanical Engineers (ASME) Nuclear Quality Assurance (NQA)-1 are available for test facility quality assurance. NQA-1 also provides criteria for assessing historical data to facilitate compliance with quality assurance requirements (ANL, 2020). Alternative approaches, such as commercial grade dedication, may also be acceptable means for justifying that data was collected under an appropriate quality assurance program. ED G3.1 can be satisfied by demonstrating that data were collected under an appropriate quality assurance program or by otherwise justifying the use of existing data.

Letter 3—Public

Comment No. 3-1

In the presented document, the ongoing impact/influence of LWR fuel requirements is being felt. It is probably very difficult to get rid of this. If this does not work and it is impossible to abstract, it is proposed to separate the LWR requirements. And even highlight them in a separate section or even in a separate document.

NRC Response

See response to Comment 2-1.

Comment No. 3-2

The document should contain a specific and precise technical definition of subject "fuel qualification" and "fuel qualification process".

NRC Response

In response to comment, the NRC staff added Section 1.2, “Definitions,” and provided definitions for “qualified fuel,” and “fuel qualification.”

Comment No. 3-3

Chapter 1.2.

Such document should not include phraseology like "for example". in addition, such an official document can't include references to specific types of fuel even as an example. Such

references could be incorrectly interpreted by applicants and allow them, in turn, to refer to this official document as on recommendation.

Moreover, there is sufficient scientific, engineering, technical evidence that fuel and core should be considered exclusively in conjunction with PCS and all complex of reactor unit safety systems. This means that fuel can't qualify separately from PCS and/or safety systems. Probably this does not need to be specified strictly, but a more specific reference in the document is necessary.

NRC Response

The NRC staff disagrees with this comment because it is standard practice to provide clarifying examples, either directly or by reference, in guidance.

Regarding the consideration of fuel in conjunction with reactor safety systems, NUREG-2246 addresses this consideration in the following sections:

1. Section 2.1, “Regulation Basis” states:

   Note that satisfying the fuel qualification framework criteria only “partially addresses” the requirements associated with the nuclear facility. This is because the fuel qualification framework provides a means to identify the safety criteria for the fuel and it is the safety criteria for the fuel that establish the performance criteria for some structures, systems, and components (SSCs) of the facility. Therefore, addressing the criteria in the fuel qualification framework provides the information necessary to meet the regulations, but does not in and of itself satisfy regulatory requirements. The requirements are fully addressed through the description and analysis of these SSCs in an application.

2. Section 3.2.1.1 G2.1.1 – Definition of Fuel Performance Envelope

   The envelope may be specified by fuel designers and may constrain the design of the reactor and associated systems. Alternatively, a reactor design may be proposed that imposes constraints on fuel performance.

Additionally, NRC staff experience reviewing EPRI-AR-1(NP)-A, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance,” (ADAMS Accession No. ML20336A052) demonstrates that regulatory findings associated with a nuclear fuel can be made without reference to a specific reactor design. The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 3-4

Chapter 1.3.

A more detailed enumeration of the neutron-physical characteristics is probably required, not only mentioned "feedback". Also, the phrase "thermal-fluid performance" here does not sound like a strictly exact technical definition in context.

It is also unclear how fuel (itself) affects the reactor core seismic behavior if the topic here is rather a fuel composition (fuel meat if you like) but not fuel design. Probably it is necessary to separate and/or clarify the concepts of "fuel" as fuel composition, and/or "fuel", as fuel elements design or even "fuel" as core (itself).

<u>NRC Response</u>

The NRC staff does not believe it is necessary to enumerate the items that are outside the scope of the report.

Regarding the comment on reactor core seismic behavior, please see NUREG-0800, Section 4.2, Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," (ADAMS Accession No. ML070740002) for a clarifying example.

The NRC staff made no changes to NUREG-2246 in response to this comment.

<u>Comment No. 3-5</u>

Chapter 2.1. etc. (probably required additional consideration)

Such document should not include phraseology like "note that ...", or "some", or "reviewed multiple times".

Probably clarification of the term "nuclear facility" (vague here) is required, and a different term should be used. The facility, in this case, is not an accurate term to define a reactor facility. And fuel, first of all, qualifies for specific reactor design/type usage.

When referring to "8 criteria" it is necessary to put a reference on the original document. It seems to me that this paragraph should be defined more precisely, although I quite understand what it is about.

Chapter 2 has many useful links (references), but all these links are too complicated, not structured, and not built in a systematic way. I would advise you to build a table or diagram with the connections of points and references for additional information. Such a diagram will immediately show unnecessary intersections and inaccuracies.

<u>NRC Response</u>

The NRC staff disagrees with the comment regarding the use of clarifying language and examples (e.g., "note that" and "some") because it is standard practice to provide clarifying language and examples in guidance.

The NRC staff believes that the use of the term "nuclear facility" is accurate in this NUREG. However, NUREG-2246 will be updated to replace "nuclear facility" with "nuclear reactor," where appropriate.

The comment regarding "8 criteria" appears to be out-of-scope because this term is not used in NUREG-2246.

The related guidance in Section 2.2 were not originally developed systematically to interface with NUREG-2246. However, there are relationships between the referenced guidance and

NUREG-2246 (e.g., lessons-learned from past guidance, concurrent NRC activity that was considered in the development of NUREG-2246) and internal NRC reviews identified the need to clarify these relationships.

The NRC staff replaced “nuclear facility” with “nuclear reactor” in NUREG-2246 in response to this comment.

Comment No. 3-6

Chapter 3. etc.

The main goal decomposition does not seem correct (Fig. 3.1). G1 and G2 not only goals for fuel qualifications. The manufacturing process and safety criteria vs. qualification? Truly not sure about it. Can you formalize, which “fabrication parameters significantly affect” what? At the same time, below “complete manufacturing specification is not expected as part of the licensing documentation”…

Also, I am not exactly confident about the correctness of this statement… “reactivity-initiated accidents for LWRs with zirconium-based cladding” it's pure Zr chemical property, not reactivity-initiated. For example, Cr-Ni high-temperature alloy response absolutely differently during the same, so-called “reactivity-initiated accident”. Let’s me ask if the temperature of Zr-cladding will increase during the residual heat removal process, because of coolant loss, what kind of accident is this? Also would be reactivity initiated?

I have recreated the full diagram for subsections 3.1. and 3.2. See the figure for notes. Probably easy to understand author's ideas with this diagram. I couldn't understand why the authors use the term safety criteria instead of fuel acceptance criteria? Safety criteria for fuel connotations require a more precise definition, which will help the authors to clearly understand that this term, applied separately to fuel, does not seem correct.

Sections 3.3 and 3.4 do not contain anything new for fuel qualification process. Nothing from these sections would directly affect the modeling of fuel behavior and experiments with fuel. Any applicable and/or developed models, and any experiments in critical areas are evaluated according to approximately the same algorithms, techniques, procedures. And these procedures required justification separately, and definitely not in this document.

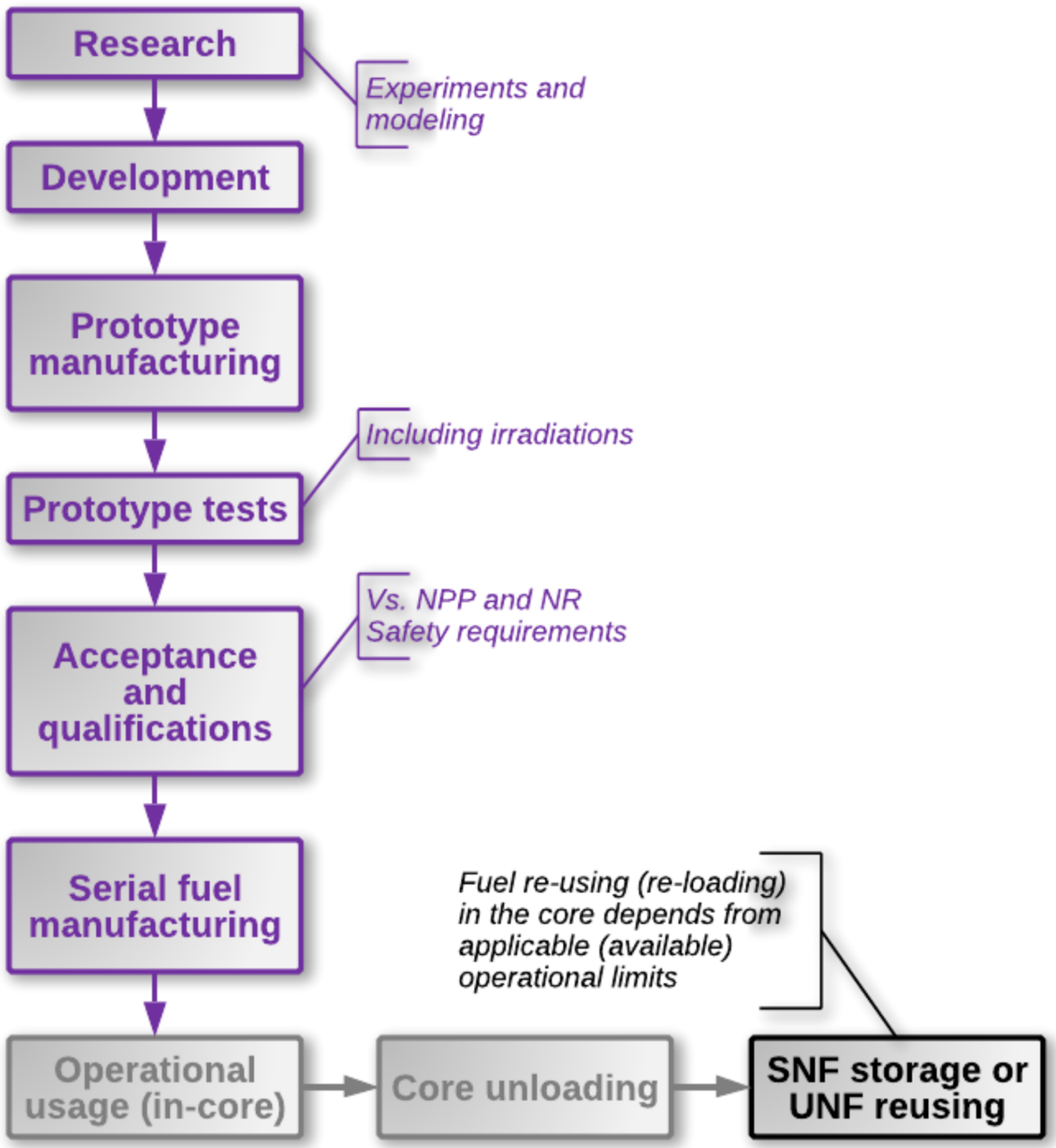
If subsections 3.3 and 3.4 should contain only recommendations specifically related only to the NR fuel qualification process and specified for this process exclusively, this will significantly reduce the disagreement between the developer and the licensee institute (NRC). Overwise risk of endless discussions increasing.

That is, all descriptions, diagrams, and tables A-2 and A-3 do not make much sense and not related to fuel qualification process.

In Table A-1, all block names must be clearly identified with the diagrams and should be the same. A single large diagram will be more useful than separate blocks of it because this diagram could present the logic of the process.

I am working on logic now and will present it my remarks additionally. I understand your logic, but this logic does not look perfect for such important document. Diagram block G.2.1.1 has connections to two different blocks (for example). But these blocks are different in nature and

cannot be substantiated simultaneously. Otherwise, this is not correct from the technical point of view.



NRC Response

Regarding the composition of the framework, the framework has received input from international regulatory safety authorities, undergone internal NRC review, been subject to public stakeholder meetings and a meeting with the Future Plant Designs Subcommittee of the NRC's Advisory Committee on Reactor Safeguards. The NRC staff does not have sufficient information to generically describe "which fabrication parameters significantly affect what".

Details and clarifying examples are provided in Section 3.1, “G1 – Fuel Manufacturing Specification,” of NUREG-2246.

Regarding the correctness of the statement, “failure criteria during reactivity-initiated accidents for LWRs with zirconium-based cladding depend upon the heat treatment of the cladding because of its impact on microstructure,” NRC staff confirms the correctness of the statement. See NRC Regulatory Guide 1.236, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents,” (ADAMS Accession No. ML20055F490) and its technical basis document “Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1,” March 16, 2015 (ADAMS Accession No. ML14188C423) for detailed clarification.

Regarding the questions on cladding heatup during residual heat removal, the specific physics modeling needed to address this scenario and the scenario classification for this event are outside the scope of NUREG-2246.

The NRC staff disagrees with the comments regarding Section 3.3, “Assessment Framework for Evaluation Models,” and Section 3.4, “Assessment Framework for Experimental Data,” because these sections address considerations that the NRC staff have identified as important to support licensing actions associated with nuclear fuel (e.g., a topical report on fuel qualification, a reactor licensing submittal that references a novel fuel design). This information currently does not exist in an NRC guidance document.

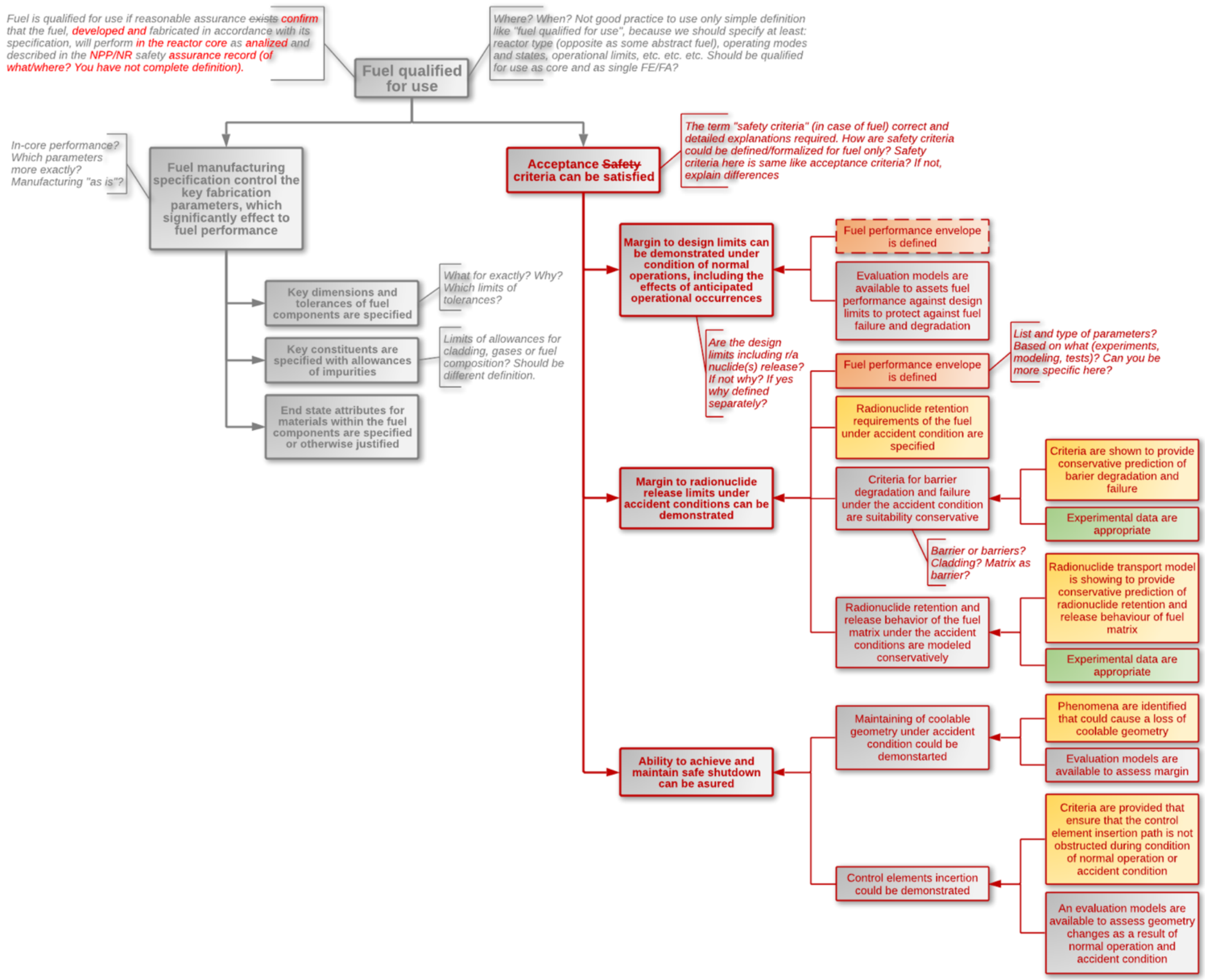
Regarding the listing in Table A-1, “List of Goals in Fuel Qualification Assessment Framework,” NRC staff paraphrased the complete description of some goals for brevity.

The NRC staff made no changes to NUREG-2246 in response to this comment.

Comment No. 3-7

Additional remarks about your document would be presented on diagram (see below).

In addition, a diagram of the fuel creation process could be useful, for example, like the one below, this diagram reflects the process of fuel creation and qualification for LWRs. But this diagram been created it after the fact. And today, the fuel development processes prediction are required, "a look beyond the horizon."



NRC Response

The NRC staff interprets many aspects of this comment to be design specific and the NRC staff does not have sufficient information to provide example criteria for specific advanced reactor fuel types at this time (see response to Comment 2-1).

Regarding the comment about the diagram reflecting the fuel qualification of LWRs, NRC staff acknowledges that the framework incorporates lessons-learned from licensing LWR fuel. The assessment framework draws on regulatory experience gained from licensing solid fuel reactor designs (particularly LWR designs), results from advanced reactor fuel testing performed to-date, and accelerated fuel qualification considerations. An attempt has been made to develop a generically applicable set of base goals. However, some goals may not apply to all fuel types or reactor designs.

The commenter’s suggestion to create a diagram of the fuel creation process is outside the scope of this report, and no changes were made to NUREG-2246 in response to this comment.

Comment No. 3-8

Additionally:

In the document: “The purpose of this report is to provide a fuel qualification assessment framework for use with advanced reactor designs that satisfies regulatory requirements”.

First of all, you need to accurately and correctly answer the question what for this document been developed? Your statement does not look good enough. Where are such “regulatory requirements” for fuel (different types) formalized officially?

At least half of problem such correct formalization. Which document fuel developers can see? Where is this document in references?

In the document: The objective of nuclear fuel qualification is to: “demonstrate that a fuel product fabricated in accordance with a specification behaves as assumed or described in the applicable licensing safety case, and with the reliability necessary for economic operation of the reactor plant”

This is not perfect definition, probably good for 2007, but not perfect for today. How the economic related to so called "advanced reactor design"? We have no real economical estimation, evidence and scientifically accepted and justified analysis to say something like this. You should remember, this is not presentation for investors, this is real technical document.

Based on presented information, document authors considering fuel qualification as fuel manufacturing "as is" (only) vs. safety criteria (see diagram). At the same time, authors trying to give advises about modeling and research (experiments) approach (mentioned in my remarks above). Not seems correct approach.

In the document: Nuclear fuel affects many aspects of nuclear safety, including neutronic performance (e.g., reactivity feedback), thermal-fluid performance (e.g., margin to critical heat flux limits), fuel mechanical performance, reactor core seismic behavior, fuel transportation, and storage.

But, CHF limits good for LWR, and authors trying to separate LWR and advanced reactors in the document. I do not see thermal-physical fuel performance, parameters (mechanical parameters)? How fuel involve to seismic behavior (you considering fuel, not core end not even fuel elements or fuel assembles) if you not considering structural analysis and fuel design? Any comments here?

NRC Response

The NRC staff added text to Section 1.1 of the report to address the comment related to the purpose of this report. This revision also eliminated the quotation that stated the objective of nuclear fuel qualification (originally taken form the 2007 paper by Crawford, Porter, Hayes, Meyer, Petti, and Pasamehmetoglu).This quotation was identified as potentially misleading.

Regarding the comments on evaluation models and experimental data, Section 3.3, “Assessment Framework for Evaluation Models,” and Section 3,4, “Assessment Framework for Experimental Data,” of NUREG-2246 address considerations that the NRC staff have identified as important to support licensing actions associated with nuclear fuel (e.g., a topical report on fuel qualification, a reactor licensing submittal that references a novel fuel design). This information currently does not exist in an NRC guidance document. This item was also addressed in response to comment 3-6.

Regarding the comment on reactor core seismic behavior, please see NUREG-0800, Section 4.2, Appendix A, “Evaluation of Fuel Assembly Structural Response to Externally Applied Forces,” (ADAMS Accession No. ML070740002) for a clarifying example.

The remaining comments appear to be outside the scope of NUREG-2246.

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11. ABSTRACT (200 words or less)

Proposed advanced reactor technologies~~designs~~ use fuel designs and operating environments (e.g., neutron energy spectra, fuel temperatures, neighboring materials) that differ from traditional light-water reactor fuel for which there is a considerable~~the large~~ experience base~~ available for traditional light-water reactor fuel~~. The purpose of this report is to identify criteria that will be useful for advanced reactor designers through an assessment framework that would support regulatory findings associated with nuclear fuel qualification. The report ~~begins by ~~examines~~ing~~ the regulatory basis and related guidance applicable to fuel qualification, noting that the role of nuclear fuel in the protection against the release of radioactivity for a nuclear ~~facility~~ reactor depends heavily on the reactor design. The report considers the use of accelerated fuel qualification techniques and lead test specimen programs that may shorten the timeline for qualifying fuel for use in a nuclear reactor at the desired parameters (e.g., burnup). The assessment framework particularly emphasizes the identification of key fuel manufacturing parameters, the specification of a fuel performance envelope to inform testing requirements, the use of evaluation models in the fuel qualification process, and the assessment of the experimental data used to develop and validate evaluation models and empirical safety criteria.

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